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# Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components

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Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

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# Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components

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## ABSTRACT

This report comprises Volume 1 of a review of U.S. nuclear plant regulatory instruments to determine the amount and kind of information they contain on managing the aging of safety-related components in U.S. nuclear power plants. The review was conducted for the U.S. Nuclear Regulatory Commission (NRC) by the Pacific Northwest Laboratory (PNL) under the NRC Nuclear Plant Aging Research (NPAR) Program. Eight selected regulatory instruments, e.g., NRC Regulatory Guides and the Code of Federal Regulations, were reviewed for safety-related information on five selected components: reactor pressure vessels, steam generators, primary piping, pressurizers, and emergency diesel generators. Volume 2 will be concluded in FY 1991 and will also cover selected major safety-related components, e.g., pumps, valves and cables. The focus of the review was on 26 NPAR-defined safety-related aging issues, including examination, inspection, and maintenance and repair; excessive/harsh testing; and irradiation embrittlement. The major conclusion of the review is that safety-related regulatory instruments do provide implicit guidance for aging management, but include little explicit guidance. The major recommendation is that the instruments be revised or augmented to explicitly address the management of aging.

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# PREFACE

This report was developed to provide a preliminary assessment regarding the extent of those regulatory instruments that contain information pertinent to managing aging. Assessments of the applicable regulatory instruments regarding aging management is complex and subject to differences in interpretation. Therefore, the perspectives in this report should be considered preliminary. These perspectives are not established needs or views and do not reflect regulatory positions or requirements.

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# ACRONYMS

| | |
|---|---|
| A&E | Architect Engineer |
| ACI | American Concrete Institute |
| ANL | Argonne National Laboratory |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute, Inc. |
| ASME | American Society of Mechanical Engineers |
| ASTM | American Society of Testing and Materials |
| BPVC | Boiler and Pressure Vessel Code |
| BTP | Branch Technical Position |
| BWR | Boiling Water Reactor |
| CCS | component, systems or structure |
| CFR | Code of Federal Regulations |
| DMS | diminishing manufacturing source |
| EDG | emergency diesel generator |
| EMTB | Material Engineering Branch |
| EPRI | Electric Power Research Institute |
| FSAR | Final Safety Analysis Report |
| GDC | General Design Criteria |
| GSI | Generic Safety Issues |
| HAZ | heat affected zone |
| HPI | high pressure injection |
| IEEE | Institute of Electrical and Electronic Engineers |
| IGSCC | intergranular stress corrosion cracking |
| ISI | inservice inspection |

| | |
|---|---|
| LE | life extension |
| NDE | nondestructive examination |
| NDT | nondestructive testing |
| NFC | National Fire Code |
| NPAR | Nuclear Plant Aging Research |
| NPP | nuclear power plant |
| NRC | Nuclear Regulatory Commission |
| NSSS | nuclear steam supply system |
| NUMARC | Nuclear Management and Resources Council |
| NUPLEX | Nuclear Plant Life Extension |
| NUREG | Nuclear Regulatory Commission Report |
| PLEX | Plant Life Extension |
| PNL | Pacific Northwest Laboratory |
| P/T | pressure/temperature |
| PTS | pressurized thermal shock |
| PVRC | Pressure Vessel Research Committee |
| PWR | pressurized water reactor |
| QA | quality assurance |
| RCC | reactor core cooling |
| RCPB | reactor coolant pressure boundary |
| RCS | reactor coolant system |
| RG | NRC Regulatory Guides |
| RHR | residual heat removal |
| RPV | reactor pressure vessel |
| $RT_{(NDT)}$ | reference nil-ductility temperature |
| SAR | Safety Analysis Report |

| | |
|---|---|
| SCC | stress corrosion cracking |
| SG | steam generator |
| SRP | Standard Review Plan |
| SSE | safe shutdown earthquake |
| SWG | Special Working Group |
| TDI | Transamerican DeLaval Inc. |
| TS | Technical Specification |

## SUMMARY

This report comprises Volume 1 of a review of U.S. nuclear plant regulatory instruments to determine the extent that they contain information pertinent to managing aging of safety-related components in nuclear power plants. The instrument review was conducted for the U.S. Nuclear Regulatory Commission (NRC) under the NRC Nuclear Plant Aging Research (NPAR) program.

As used in this report, an "instrument" is a procedure or document as in an **instrument of government**, e.g., the Code of Federal Regulations. The terms **regulatory instruments** or **instruments** are used throughout this report and in this context are not to be confused with an instrument of measurement, e.g., a pressure gauge or flow meter. Eight regulatory instruments were selected for the review:

- Code of Federal Regulations (CFR)
- Technical Specifications (TS)
- Standard Review Plan (SRP)
- NRC Regulatory Guides (RG)
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Sections III and XI
- Generic Safety Issues (GSI)
- American Nuclear Society (ANS) Standards
- Institute of Electrical and Electronics Engineers, Inc. (IEEE) Standards.

Historically, all these regulatory instruments have been used for the design, construction, start-up and operation of Nuclear Power Plants. In this, their primary purpose has been to protect the health and safety of the general public. The intent of this review was to determine the degree of emphasis on the management of aging found in the instruments now in use. No attempt was made to take into account that approximately 50% of the Nuclear Power Plants now operating were licensed before 1975. Therefore, not all of the plants were licensed under the same set or edition of instruments noted above. The review of the applicable instruments was based on the technical issues related to aging. The descriptions included are the author's and do not represent NRC considerations for license renewal.

The components on which the review focused were five light water reactor major safety-related components: reactor pressure vessels, steam generators, primary piping (reactor coolant piping), pressurizer vessel, and emergency diesel generators. (Cables, containment and basemat and selected pumps and

valves will be investigated in FYs 1990 & 1991 and will be published as Volume 2.) The components were selected from the NPAR program document NUREG-1144, Nuclear Plant Aging Research Program Plan, that has identified a list of components of current interest and concern relative to the management of aging. The components are high priority, safety-related components that have been, in varying degrees, subject to aging problems.

The focus of the review was on 26 NPAR-defined aging issues, including generic issues, e.g., examination, inspection and maintenance, embrittlement, corrosion, erosion, and thermal cycles; and component specific issues, e.g., steam generator tube specific - intergranular attack, fretting and denting, and emergency diesel generator specific - harsh and frequent testing.

Each regulatory instrument was evaluated for each aging issue chosen for each component, e.g., for the **RPV** and the **aging issue of corrosion**, each instrument was evaluated for aging features that provide implied or explicit direction in the management of corrosion. The results of the review are contained in tables in appendixes for each of the major components.

The principal conclusion is that aging management does exist in the safety-related regulatory instruments; however, the information is largely implied. The emphasis of the instruments appears to be on initial design, construction, qualification and start-up and actions to address aging problems that develop after the problem is found. Finally, it was also concluded that revisions should be made in the instruments to explicitly address aging.

The major recommendation is that the existing body of regulatory instruments should directly address aging and the management of aging. The difficulties, however, with any revisions are acknowledged, and it is recommended that a project plan for the revision process be evaluated and defined. The planning should include evaluations of ongoing NRC aging research and industry aging-related research, e.g., Electric Power Research Institute and individual utilities, and the development of a realistic time frame for implementation of the revisions.

It is also suggested that all regulatory instrument review information be installed in a computer data-based system for broader use by the NRC and others.

A number of observations outside the scope of the review were developed during the analysis of the review by PNL staff. The principal observations are 1) evaluation of component replacement methods as an aging management procedure should be addressed in the regulatory instruments for augmentation of safety and cost effectiveness; 2) aging management could be enhanced by improvement of NDE methods and inspection tools; 3) the ASME BPVC, as the major contributor to design and construction and aging management, provides a valuable contribution to the NPP industry; however, improvements in the Code are needed to address component material deterioration, design that encourages and allows for repair, replacement, and improved inservice inspection, and improvements in the time cycle for revising the Code (ASME is currently addressing hard to backfit design problems in Section III and the problems of long time cycles in Section XI); and 4) a safe and well-maintained plant, i.e., a plant with excellent maintenance methods and procedures, is likely to be a plant that is successfully managing the aging of its components and systems.

# 1.0 INTRODUCTION/PURPOSE

This report comprises Volume 1 of a review of U.S. nuclear plant regulatory instruments to determine the extent that they contain information pertinent to managing the aging of safety-related components in nuclear power plants (NPPs). The review was conducted for the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, by the Pacific Northwest Laboratory[a] under the NRC Nuclear Plant Aging Research (NPAR) Program (NRC 1987a). In conducting the review, the focus was on safety-related aging as it relates to selected safety-related components in Nuclear Power Plants (NPPs). NPAR has defined aging as "the cumulative degradation that occurs with the passage of time in a component, system of structure." Therefore, the essential elements of the review on which this report is based may be presented as follows:

REGULATORY INSTRUMENTS

SAFETY-RELATED NPP COMPONENTS ⟷ SAFETY-RELATED AGING ISSUES

As used in this report, an "instrument" is a procedure or document as in an **instrument of government**, e.g., the Code of Federal Regulations. The terms **regulatory instruments** or **instruments** are used throughout this report and in this context are not to be confused with an instrument of measurement, e.g., a pressure gauge or flow meter. The regulatory instruments reviewed for safety-related information are listed in Section 2.2. Each is described in the body of this report. Historically, these safety-related regulatory instruments have been used as the basis for the design, construction, start-up, equipment qualification, and operation of NPPs. Their primary purpose has been to protect the health and safety of the general public. This review was undertaken to determine the degree of emphasis, if any, regarding the inclusion of explicit requirements for the management of aging found in the instruments now in use. No attempt was made to take into account that approximately 50% of the NPPs now operating were licensed before 1975. Therefore, not all the plants were licensed under the same set or edition of instruments noted in Section 2.1. The review of the applicable instruments was based on the technical issues related to aging. The descriptions included are the author's and do not represent NRC considerations for license renewal.

The components on which the review focused are the reactor pressure vessels; steam generators; primary piping (reactor coolant piping); pressurizer vessel, with special emphasis on pressurizer spray and surge lines and

(a) Operated for the U.S. Department of Energy by Battelle Memorial Institute.

internals; and the emergency diesel generators. (Underway or planned are reviews for cables, containment and basemat and selected pumps and valves; this work will be published as Volume 2 in FY 1991).

To determine whether the selected instruments contain significant aging safety-related information related to the five safety-related components, the review concentrated on NPAR defined safety-related aging issues. A complete list of the 26 identified aging issues is found in Section 2.3. These aging issues include examination, inspection and maintenance, embrittlement, corrosion, erosion, and thermal cycles; and component--specific issues, e.g., steam generator tube specific--denting, fretting and crevice intergranular attack, and emergency diesel generator specific--harsh and frequent testing. Each aging issue is defined in Appendix I.[a]

The results of the review on which this report is based are contained in tables in Appendixes II[a] through VI entitled "Regulatory Instrument Reviews": Appendix II for Reactor Pressure Vessels; Appendix III for Steam Generators; Appendix IV for Primary Piping (Reactor Coolant Piping); Appendix V for Pressurizer Internals; and Appendix VI for Emergency Diesel Generators.

Section 2.0, "Selection of Instruments, Components and Issues, and Methodology," describes the regulatory instruments, the selection of components, the rationale for the aging issues, the instrument review criteria, and the review methodology. Also included in Section 2.0 are two typical examples of how to use the tabular data in Appendixes II through VI.

Section 3.0, "Discussion," provides background information on the regulatory instruments versus aging management and the aging issues. Specific examples of aging management guidance, either direct or indirect, are provided for each instrument.

The "Conclusions and Recommendations," "Suggested Future Regulatory Instrument Review Activities," and "Observations Outside the Scope of the Review" are found, respectively, in Sections 4.0, 5.0 and 6.0 of this report.

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(a) The regulatory instruments, e.g., Code of Federal Regulations and American Society of Mechanical Engineers Code, frequently make use of the Appendix A, B, etc., within their text; therefore, the appendixes attached to this report will be designated in Roman numerals to distinguish them from those found in the instruments.

## 2.0 SELECTION OF INSTRUMENTS, COMPONENTS AND ISSUES, AND METHODOLOGY

This section describes regulatory instruments, the selection of the major components, the rationale for selection of the aging issues, and the development of the review criteria and methodology. Examples of how to use the review information contained in appendixes are included.

### 2.1 REGULATORY INSTRUMENTS REVIEWED

The following regulatory instruments were reviewed for this report. While not necessarily all inclusive, they are the most important and applicable to the pressure boundary aspects of the current components. IEEE Standards were added to the list to account for the electrical aspects of the emergency diesel generators and pressurizer. IEEE Standards were not considered in the reviews of RPV, steam generator (SG), or the primary piping. PNL staff with experience in the aging issues and familiarity with the applicable regulatory instrument performed the reviews and interpreted the instrument text.

- Standard Review Plan (SRP)
- Code of Federal Regulations (CFR)
- NRC Regulatory Guides (RG)
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Sections III and XI
- Generic Safety Issues (GSI)
- Technical Specifications (TS)
- Institute of Electrical and Electronics Engineers, Inc. (IEEE) Standards
- American Nuclear Society (ANS) Standards.

Each instrument is described below.

- Standard Review Plan. The SRP is published by the NRC and provides guidance for the review of Preliminary Safety Analysis Reports (SAR) and the Final Safety Analysis Reports (FSAR) for plant design and operations. The SRP contains 18 chapters and covers all facets of NPP operations, including plant descriptions, design and construction, safety features, instrumentation, electrical power, radiation protection, waste management, quality assurance, and human factors engineering. The SRP also presents specific Branch Technical Positions (BTPs) that are developed and published by branches within the NRC, e.g., BTP ISCB 19 by the Information Security Branch found in SRP Chapter 7, Instrumentation and Controls.

- Code of Federal Regulations. These regulations codify general and permanent rules published in the U.S. Federal Register by the Executive departments and agencies of the U.S. Federal Government. With reference to this report, these codes establish the rules for design, construction licensing and operation of commercial NPPs. For the NRC, a federal agency, the most significant CFR relating to commercial reactors is Title 10, Chapter 1, Part 50, Domestic Licensing of Production and Utilization Facilities. Part 50 includes important appendixes, such as the General Design Criteria requirements, specific material requirements for reactor vessels, emergency planning, and quality assurance criteria for NPPs.

- Regulatory Guides. These guides are published by the NRC in 10 broad divisions, including Division 1, Power Reactors. The guides are available to the public and NPP licensees. They provide general guidance to applicants and describe methods acceptable to the NRC staff, to implement specific parts of the Commission's regulations and, to delineate evaluation techniques used by NRC staff for specific problems and postulated accidents. The RGs often explain and detail acceptable methods for the rules found in the CFRs.

- American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Sections III and XI. The ASME Code establishes the rules of safety governing design, construction, operation and testing of NPP components and systems. Section III covers a broad range of components and systems including, pressure vessels, piping, pumps, valves, supports and core internals. Section XI provides rules for inservice inspection of components and systems. Section XI constitutes the requirements of examinations, testing and inspection to maintain an NPP in a safe and expeditions manner. Section XI is applicable when the requirements of the Construction Code, e.g., Section III, have been satisfied. The ASME Code Editions, Addenda and Code Cases used by the owners of NPPs are subject to acceptance by the NRC. The ASME codes are frequently referenced in the CFRs. The ASME Maintenance and Operation (O&M) committees provide an important function through the testing of pumps, valves and snubbers.

- Generic Safety Issues. GSIs are published by NRC to identify safety issues generic to NPPs. The GSIs can cover a wide variety of subjects; however, for the purposes of this review, the GSIs used are those that are predominately concerned with material degradation and operations that cause degradation of plant components. When a solution to the generic issue is found, the usual result is an NRC Generic Letter or an NRC NUREG report that provides an industry wide dissemination of the solution.

- Technical Specifications. Technical Specifications are NPP mandatory operational specifications that provide instructions for limiting conditions and surveillance requirements for plant operations. License applications for authority to operate an NPP are required to supply Technical Specifications as enumerated in 10 CFR 50, Section 50.36.

- Institute of Electrical and Electronic Engineers Standards. The IEEE Standards present criteria and requirements for electrical systems that are specifically related to providing protection to the health and safety of the public. The standards included in this review are principally those prepared by the IEEE Nuclear Power Engineering Committee. The IEEE Standards, through the Equipment Qualification (EQ) standards define "qualified life" and provide guidance for requalification or replacement of components when their qualified life is reached. The IEEE standards are not mandatory and use of the standards is wholly voluntary. Use of the standards, as acceptable practice, are subject to the approval of the regulatory agency, i.e., the NRC. Specific IEEE standards are frequently called out and/or qualified in the Regulatory Position Section of the RGs as acceptable to NRC staff.

- American Nuclear Society Standards. The ANS standards are agreements among designers, engineers, governmental regulatory agencies, manufacturers, and nuclear scientists. The standards are developed to provide current practices on various subjects that affect NPPs. Among the subjects are criteria for earthquake instrumentation, various safety guides, selection and training of personnel, QA, security for NPPs, and auxiliary feedwater system for PWRs. The ANS standards are frequently published as American National Standards with the approval of the American National Standards Institute (ANSI). ANS standards are guides for prospective use and are not mandatory; the designers are not restricted to the ANS standards and may propose alternate criteria to provide adequate safety.

These instruments are listed in column 2 of the component reviews Appendixes II through VI.

## 2.2 MAJOR COMPONENTS

The following components were chosen for the current review. They were selected from the major LWR plant elements of current interest in the NPAR program, as found in Table 5.3 of the NPAR program document NUREG-1144, Nuclear Plant Aging Research Program Plan. These selected components are recognized by the NRC as safety-related components that historically, in varying degrees, are subject to aging problems. Finally, the selected components provide a good mix for the review by covering the three major design disciplines, i.e., structural, mechanical and electrical. The components are listed in Column 1 of Appendix II through VI.

- reactor pressure vessel

- steam generator

- primary piping (reactor coolant piping)

- pressurizer
- emergency diesel generator.

## 2.3 AGING ISSUES

Listed below is a list of the component aging issues used in the instrument review.

- corrosion
- corrosion/erosion
- crack initiation and flaw propagation
- creep/swelling
- element burnout(s)
- erosion
- examination, inspection and maintenance and repair
- excessive/harsh testing
- fatigue
- fracture toughness
- intergranular stress corrosion cracking
- irradiation embrittlement
- low-flux long-time irradiation of vessel internals and supports
- seismic failure/damage
- specific environmental factors, e.g., moisture, oxidation, chemicals, oils, and dust.
- stress corrosion
- stud failure
- thermal cycles
- thermal embrittlement
- thermally induced bending

- thermally induced mechanical wear
- transient thermal loads
- tube specific - denting, fretting and crevice corrosion
- vibration
- wear
- weldments - specific to dissimilar metals of safe ends.

For illustrative purposes, some of the aging issues of the instrument review are shown below.

For the reactor pressure vessel (RPV),

- IRRADIATION EMBRITTLEMENT - degradation of the structural integrity of vessel materials due to the consequences of long-term exposure to high radiation levels that induce changes in the vessel's material properties.
- FATIGUE - degradation associated with cyclic loading, transient cycles, and thermal and pressure cycles.

For the steam generator (SG),

- CORROSION - degradation associated with the deterioration of material surfaces, chiefly, through chemical actions. The chemical reaction is influenced by environment of the material or component.
- TUBE SPECIFIC (DENTING, FRETTING AND CREVICE CORROSION) - aging issues associated with the generator tubes include denting caused by the crevice corrosion of the tube support plate and tube sheet material; fretting is caused by the wear action between the tubes, tube supports and antivibration bars; crevice corrosion due to localized stagnant solutions in lap joints, holes, welding surfaces, etc.

All specific aging issues were derived from the general guidance found in NUREG-1144, Rev. 1., Section 2.1 (NRC 1987a). Typical degradation mechanisms are neutron embrittlement, fatigue, erosion, corrosion, oxidation, thermal embrittlement, and chemical reactions. Aging is also induced by stressors, e.g., service wear, testing, improper installation, and application and maintenance.

Each aging issue was considered, and those believed by PNL staff to be specific to the components were chosen for the review. The aging issues for each component were then grouped into principal categories that define the mechanics or phenomena of the aging processes for the component. For a typical example see Appendix III, page III.9, item (50), Dynamic effects, that includes vibration, thermal cycles and erosion.

Aging issues are not necessarily specific or all inclusive for each component nor do they manifest themselves in the same way for each component. For example, neutron embrittlement is not typical to all components. Radiation is unique to the RPV and is not generally regarded as a problem in the other review components. On the other hand, the RPV as a pressure retaining component has many safety-related degradation problems that are common to all vessels and piping systems.

The aging issues for each component are found in column 3 of the review Appendixes II through VI.

## 2.4 REVIEW CRITERIA

The review criteria chosen to examine the regulatory instruments answer three questions: 1) Do the instruments address aging? 2) If so, in what form is the aging addressed? For example is aging management implied or explicit and is the guidance or direction adequate or incomplete for the aging issue? 3) What are the current initiatives that would change the instruments to address aging and life extension?

The review criteria as they appear in the component review sections are noted below with a brief definition:

1. Aging Features: What parts of the instrument, if any, identify the management of component aging. Are aging issues addressed by the instrument? Is the information implied or explicitly expressed?

2. Life Extension Features: Does the instrument address life extension or component replacement needs?

3. Current Initiatives: Is work currently underway to update or change the instrument to include aging features or life extension features? What, if any, is the status of special working groups, of technical committees, or of on-going research that could lead to the revision of the instruments to include these features?

4. Aging Needs: What changes or revisions generally are needed in the instrument to address aging issues?

5. Life Extension Needs: Does the instrument require further investigation/research of the aging issue to accommodate life extension? How could the instrument be used or modified to meet life extension needs?

The review criteria are listed in columns 4 through 8 on each page of the component review sections, Appendixes II through VI.

## 2.5 REVIEW METHODOLOGY

Each regulatory instrument was examined for data for each designated major component's aging issues. Thus the regulatory instrument review for a component with five aging issues required five reviews for each instrument. The review relative to the components was carried out in the generic sense; that is, no distinction was made for the different configurations or manufacturers of SGs or for RPVs, or for PWRs versus BWRs. However, the review does account for the regulatory documents that specifically address a design configuration or type of reactor. The general approach was to review the instrument for reference to the aging issues or for a specific reference to the major components by name or system.

Typically, minimal specific component or aging issues references, by name, are found in the regulatory instruments. The principal reference is to systems, such as the **reactor coolant system** or **reactor containment**. However, there are some exceptions; Appendix G, in 10 CFR 50, is specific for **fracture toughness requirements** and Appendix H is specific for **surveillance requirements** to monitor changes in fracture toughness. If the instrument revealed any relationship or potential relationship to the aging issue, further analysis was conducted to determine the explicit and/or implied relationship. The related information, if found, was analyzed by using the review criteria of Section 2.4. Applicable results of the five review criteria analysis were entered in columns 4 through 8 for each of the aging issues designated for each component (see Appendix II through VI). Special attention was given to the current initiatives criteria, and a number of on-going actions were noted throughout the review. Criteria 1, Aging Feature, is the key criteria of this review and provides the bulk of the information found in the regulatory instruments and indeed provides the emphasis to examine the instrument in more detail. The larger instruments, e.g., the SRP and ASME code, were investigated by subsections, chapters, etc., and those sections that were completely unrelated to the aging issues were ignored for detailed analysis. The smaller instruments, e.g, RGs and IEEE standards, were investigated as a whole. Many of the instruments were eliminated by title or subject or objectively determined not to be related to the aging issue(s).

## 2.6 REVIEW EXAMPLES

Two examples from the PNL regulatory instrument review are shown below. Example 1 examines the NRC RGs for the issue of EXAMINATION, INSPECTION AND MAINTENANCE AND REPAIR as they apply to the RPVs. Example 2 examines the GSIs for the issue of IRRADIATION EMBRITTLEMENT as applied to the RPVs. Each example begins with a question, followed by an answer.

### Example 1

Do the RGs refer to the aging issue of EXAMINATION, INSPECTION AND MAINTENANCE AND REPAIR with respect to RPVs?

- On page II.5 of Appendix II, five entries are shown for the RGs items (15) through (19). For example, item (15), RG 1.150, indicates that this RG does have an Aging Feature (Column 4), "UT of RPV welds," discusses evaluation of prior UT examinations for determination of crack growth rates, and Life Extension Features (Column 5), "Inservice inspection (ISI)," discusses continued structural integrity of reactor through reliable flaw detection vessels. No Current Initiatives (Column 6) were found; the entry refers to Note 1, page II.3 which indicates a need for further study. Aging Needs and Life Extension Needs (Columns 7 and 8) were shown to be suggestions for improved flaw detection and assessment. The comments (Column 9) note that cracks may propagate from cladding into the steel (shell) in the RPV and raises the question that 100% UT may be required for BWR to assess under cladding cracks.

<u>Example 2</u>

Do the GSIs refer to the aging issue of FRACTURE TOUGHNESS with respect to RPVs?

- As shown on page II.10 of Appendix II, three entries, items (56) through (58), were made for the GSI relative to the noted issue. The first entry, Item (56), indicates that GSI A-11 (Column 3) does have an Aging Feature (Column 4). Column 4 notes that "As plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial margins of safety." The GSI does not discuss Life Extension Features (Column 5) and it has no Current Initiatives (Column 6). For the Aging Needs (Column 7) and Life Extension Needs (Column 8), an entry is made indicating that further analysis is needed to resolve these review criteria. The comment (Column 9) indicates the priority for this issue has been resolved with the issuance of NUREG-0744 (NRC 1982).

## 3.0 DISCUSSION

Although the review data on regulatory instruments developed in this study, as presented in the Appendixes II through VI for each component, appear to be comprehensive, the reader is cautioned that the magnitude of the literature, research time, and funding levels did not permit an exhaustive study of all the text, references and background documentation on each instrument. The review is, however, a good overview of the degree to which aging guidance is found in the regulatory instruments. The following sections provide discussion and insights as to the effectiveness of the safety regulations to provide guidance for the management of aging.

### 3.1 REGULATORY INSTRUMENTS AND AGING MANAGEMENT

The PNL review revealed that, in effect, the regulatory instruments do contain information that does relate either directly or indirectly to current safety-related concerns and aging management. A strong corollary exists between safety and aging management, i.e., it is implied that a safe plant is more likely to be well maintained and more effectively monitored for problems and degradation. Yet, generally speaking, the instrument review does not reveal explicit requirements for aging and life extension needs or features. The majority of related information, i.e., the aging features of the instruments, is interpreted by PNL staff to be implied guidance. The implied guidance or, in some cases mandatory guidance, i.e., the technical specifications (TSs), have been useful in the management of aging. Simply stated, the safety-oriented regulatory instruments used for NPP have indirectly managed aging through safety-oriented design and safe operations.

### 3.2 REGULATORY INSTRUMENTS AND THE AGING ISSUES

This section presents the key objective of the review, which is contained in the question, "Do the regulatory instruments address aging issues?" In this section, a general assessment statement is provided for each instrument relative to the aging issues.

Each issue is listed in Section 2.3 and defined in the glossary of Appendix I. Examples of specific features in each regulatory instrument that address aging issues are included with a typical degree of depth on how the instrument addresses an issue. (In this section, the instruments are discussed in the order that they appear in the review data, Appendix II through VI.)

#### 3.2.1 Code of Federal Regulations

The CFRs are the principal base for all the regulations that mandate the design, construction, inspection and operation of the components chosen in this review. Appendix A, General Design Criteria for Nuclear Power Plants (GDC), 10 CFR 50, provides the largest measure of implied aging management.

The GDC of Appendix A coupled with Section 50.55a, Codes and Standards, provides the broad base design and inspection requirements.

The CFRs tend to be generic in description because they often address broad plant categories such as the **reactor coolant pressure boundary**. In this review, four of the components considered are within the reactor coolant pressure boundary. Consequently, the Criterion needs given in 10 CFR 50, Appendix A, implies aging management even though the component is not mentioned in specific description. In general, the review components, with the exception of the RPV, as referenced in 10 CFR 50, Appendix G and H, are not discussed in the CFRs. Aging and life extension are not explicitly cited in the text. NPAR aging, as defined in the introduction, does not appear. The implied aging features as found in the review, are based principally on the assumption that a reactor designed, constructed, tested and operated within the rules of 10 CFR 50 will provide the licensee a facility that can be managed for aging. **Aging** does appear in 10 CFR Part 50, §50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," (e), (5). This aging is in reference to equipment qualified by test and preconditioned by natural or accelerated aging. Also, in 10 CFR, Part 50, Section 50.109, Backfitting, c, (8), states that "the potential impact of difference in facility type, design or **age** on the relevancy and practicality of the proposed backfit," will be considered when addressing backfit requirements. Yet in these examples, the "aging" and "age" are not used in the same context as NPAR defined aging.

The following are typical examples of the more specific requirements of CFRs; while not using the terms aging and life extension, these examples may be regarded as aging management guidance:

- 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 51, states that "The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws." Aging and life extension are not specifically used in the above quote; however, the criterion does address a number of aging issues identified in this review including, irradiation and thermal embrittlement, fracture toughness, fatigue, thermal cycles, crack formation and growth, testing and maintenance and seismic failure/damage. The criterion terminology of "under operating, maintenance, testing, and postulated accident conditions" encompasses all plant operating phases and strongly advocates a design that assures material reliability throughout the plant's operating period.

- Appendix A, Criterion 2, address the aging issue of seismic failure/damage. The criterion states that "structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions." The design is subject to the most severe (historically) known natural occurrence, to the combinations of normal conditions and accident conditions and natural phenomena and, to the importance of the safety function performed.

### 3.2.2 Technical Specifications

The TSs are mandatory requirements that provide the safety limits, the limiting safety systems settings, the limiting conditions for operations and the surveillance requirements for NPPs. TSs should not be confused with or called "guides" because they provide specific mandatory rules for operation of a particular plant. They are required for NPP operations by the rules of the CFRs and are enforced for the life of the plant. TSs may provide aging management through enforcement of limits that are set at acceptable levels or procedures which will control, reduce, mitigate, detect or preclude aging degradation. The TSs, for example, require that safety-related equipment be in acceptable operable condition or, if not repaired within a specific time frame, the plant will be shut down. TSs also require records to be maintained for operating cycles, e.g., cyclic loading/P/T cycles. These records can provide documentation for material life or life extension documentation. The TSs reference the ASME code, Section XI, for inservice inspection.

The following are examples of specific TS applications that contain implied aging management.

- Technical specifications require inspection for "wall penetrations" or other modes of SG tube wall thinning. This requirement addresses the aging issue of erosion. The present inspection technique is eddy current for a fraction of the SG tubes on a 12-to-24 month basis.

- Thermal pressure cycles are recorded during thermal/pressure reactor coolant heat-up or cool-down on the primary side only. These cycles are useful for aging management by providing records for evaluation of the reactor coolant system P/T and PTS history.

- TSs require primary reactor coolant system (RCS) chemistry control and the retention of water analyses and associated records. This information is useful in determining the water chemistry history of RCS and is useful in the management of aging and life extension.

- The TSs require inspection of component and equipment mechanical restraints, snubbers, on a periodic basis. Failure of visual or functional tests require maintenance or replacement.

### 3.2.3 Standard Review Plan

The SRP by definition (see Section 2.1) provides a plan "for the review of safety analysis reports (SARs) for nuclear power plants." The SRP consists of eighteen chapters, ranging from site description of the plant and design of structures and components to human factors engineering. The principal concern of the SRP is safety, e.g., does the SRP review of the FSAR indicate that a plant can be started and operated safely and, if a failure occurs, can the plant be shut down in a safe manner? The SRP, typically, wants to know whether the plant design/procedures comply with applicable CFRs, such as 10 CFR 50.55a, inservice inspection. Many of the implied aging features in the SRP occur because of references to the ASME Code, 10 CFR 50, General Design Criteria and other standards.

Specific examples of aging related guidance in the SRP are as follows:

- SRP 5.2.3, Reactor Coolant Pressure Boundary Materials, discusses the review of suitable materials and review of procedures for manufacturing and welding. The SRP has implied aging management by determining whether the plant design has used the right materials, i.e., correct materials will or should stand up to the plant operation and environment.

- SRP 3.9.2 has an aging feature for the dynamic effects of vibrations through the review of the structural and functional integrity of the piping systems under vibratory loads.

### 3.2.4 Regulatory Guides

The RGs are principally issued to describe methods acceptable to NRC staff and to provide guidance to applicants in the use of the 10 CFR 50 regulations, especially as applied to the GDC. The "Introduction" of a RG will frequently refer to the importance of safety and safety systems; "aging," on-the-other-hand, is not part of the text. Yet some of the RGs do discuss the aging issues. Information on corrosion, vibration, progressive tube deterioration, and secondary water chemistry maintenance, e.g., may be found in the "Discussion" section of RG 1.83; however, the purpose of RG 1.83 is for guidance in the performance of inservice inspections (ISI) for steam generators. Some individual RGs may be specific to a single component or part of a component, and these individual RGs tend to support the direct management of aging, even though the intent of an RG is to promote safety.

Examples of aging management guidance in RGs are shown below:

- Degradation of steam generator tubes has been identified as a problem because of mechanical damage or progressive deterioration caused by inadequate design, manufacturing errors, or chemical imbalance. Tube problems are caused by a number of mechanisms including, IGSCC, IGA, pitting, denting, fatigue, wastage and erosion/corrosion. Management of the tube degradation is assisted by RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." The guide describes methods acceptable to the NRC staff for establishing limiting safe conditions of SG tube degradation. Conditions beyond the limiting safe conditions should result in removal from service by plugging (welding each end) of the tubes.

- Regulatory Guide 1.85 allows the use of approved ASME "Code Cases," thus providing a mechanism to use current repair and replacement techniques within the jurisdiction of the code. These alternate methods within the jurisdiction of the code may provide improvements in repairs and materials that could enhance aging management and life extension.

3.2.5 ASME Boiler and Pressure Vessel

The ASME Boiler and Pressure Vessel Code, especially Section III and Section XI, are used for the design and construction of NPPs. Section XI, inservice inspection remains in effect throughout the life of a plant and provides inspection criteria, intervals, and acceptable methods. Sections of the ASME code are mandated in 10 CFR 50, Section §50.55a, Codes and Standards, which specifically notes that "Structures, systems, and components of boiling water and pressurized water-cooled nuclear power plants must meet the requirements of the ASME Boiler and Pressure Vessel Code." Manufacturers and designers are expected to use sound engineering practice and judgment within the rules of the Code. The expressed purpose of the Code is to provide protection of life and property and a margin for deterioration in service for a reasonably long, safe period of usefulness. The review, however, reveals that material deterioration is not fully addressed and needs attention. The ASME Code does, in general, address "life of plant" issues and, in some cases, it is very specific, e.g., corrosion allowances for vessel and pipe thickness. The Code also provides for use of current technology for repairs/ replacement and materials by the use of approved Code Cases. The Code is a living document which is currently revised and published every three years, e.g., the latest was published July 1, 1989. Addenda are issued in December of the years between major publications. Code interpretations are usually provided twice a year as an update service to the various sections.

The following are examples of the Code that are related to aging-management issues:

- A strong implied aging features exist in Section III, Class I, piping. If corrosion or erosion is expected, the piping thickness, shall be increased over the design thickness to be consistent with the specified design life.

- The ASME Code recognizes that problems exist and, consequently, the code is under continuous revision to effect improvements and changes. For example, a Section XI, Special Working Group (SWG), has implemented revisions to IWA 2400 which will delete the current 40-year operating limit of nuclear plants. A SWG has been set up to consider the development of a new Section XI, Article IWX-8000, called Requalification of Fatigue Life. A Section III, SWG is also reviewing fatigue curves to determine revisions to accommodate operation beyond the 40 years. A new

Subsection IWT for Section XI is now under consideration by ASME; the subsection addresses the aging-related issue of erosion-corrosion.

### 3.2.6 Generic Safety Issues

Generic Safety Issues, as their name implies, are concerned with safety issues common to all NPPs or to types of LWRs, PWRs or BWRs. The GSIs do not explicitly reference aging management. The GSIs are, however, concerned with many of the aging issues, and the reader will note that, in the reviews, Appendix II through VI, the aging issues for each component are frequently the subject of a GSI, including cracking and degradation of bolts, mitigation of steam generator rupture/degradation, SCC, fracture toughness, corrosion of interior metal surfaces, control of overfilling transients, pressurized thermal shock, and neutron irradiation of RPV welds. Thus the GSIs identify many aging issues as problems, and this identification process does assist in the management of aging. When a solution to the problem is determined, management of the problem, i.e., the aging issue, is addressed by the NRC usually through Generic Letters, NUREGs and the SRP/BTP.

Examples of applicable aging issue subjects found in GSIs are listed below.

- GSI A-11 notes that as plants accumulate increasing service time that neutron irradiation reduces the fracture toughness and the initial margins of safety. This issue was thought to have been resolved by NUREG-0744 (NRC-1982); however, a need to monitor and analyze trends of nil-ductility temperature exists; that is, what are the impacts of extended life on brittleness? A related GSI is USI A-49. This issue is concerned with irradiation of reactor vessel weld and plate materials and the reduction of fracture toughness in these materials.

- GSI 29 addresses cracking and degradation of bolts and studs from stress corrosion, fatigue, and boric acid corrosion.

- GSI A-15 addresses the aging issue of corrosion activated by neutron flux. Operation of LWRs result in slow corrosion of interior metal surfaces of the primary coolant system.

- GSI A-12 addressed the problems of lamellar tearing and low fracture toughness in SG and reactor pump supports. This problem was also addressed in NUREG-0577 (NRC-1979), Potential for Gas Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports.

- GSI A-3 and A-4 reported the SG tubing degradation due to wastage and SCC in Westinghouse SG (A-3) and Combustion Engineering SG (A-4).

### 3.2.7 Institute of Electrical and Electronic Engineers Standards

Many of the IEEE standards provide qualification/acceptance criteria; thus they tend to dictate an aging management perspective through qualification testing. The IEEE standards cover a multiplicity of subjects related to electrical and electronic systems. Numerous IEEE standards are dedicated specifically to the nuclear industry's Class IE equipment, and it is these specific standards that this review investigated. The IEEE standards are frequently referenced in the regulatory instruments, such as the SRP. The regulatory basis for the use of the IEEE standards is found in the CFRs and the RGs. The NRC RGs frequently, in the "Regulatory Position" section, list the IEEE standards that are acceptable (with exceptions) to the NRC staff for satisfying the Commission's regulatory requirements. The standards are generally not acceptable in whole and the "Regulatory Position" will state the exceptions or other requirements. The majority of IEEE standards are jointly approved by the ANSI and may be designated as ANSI/IEEE and published as an American National Standard.

The following are some specific examples of the explicit and implied aging management found in the IEEE standards:

- The IEEE standards require that an assessment of the effects of aging must be addressed when considering the ability of equipment to perform safety functions. Types of aging that must be addressed include vibration, wear, and natural and environmental conditions. Standard 323 recognizes the need for aging management and defines the steps to address aging. This standard defines aging and explicitly address the subject. IEEE 323 defines aging as "the effect of operational, environmental, and system conditions on equipment during a period of time up to, but not including design basis events, or the process of simulating these events."

- IEEE standards recommend establishment of procedural practices to obtain the qualifying (test) data. The data is required to qualify that the equipment will meet its performance requirements following one safe shutdown earthquake (SSE). Included are tests for vibrational aging, seismic aging and normal operating loads. IEEE 323 defines qualification as "the generation and maintenance of evidence to ensure that the equipment will operate on demand to meet the system performance requirements."

- IEEE 934 standard addresses an aging perspective by providing criteria for the use of replacement parts for both construction and operations. The standard requires acceptance testing of parts for wear, fatigue, defects and insulation breakdown prior to release to service. Failure of parts provides aging management by addressing the aging issues, e.g., wear and fatigue and insulation breakdown.

### 3.2.8 American Nuclear Society Standards

The ANS prepares and publishes standards for the design and operation of NPPs. The standards are usually published as joint American National Standards Institute (ANSI) and ANS standards. The ANSI/ANS standards tend to be more safety specific than are the other instruments included in this review. The ANS standards do not, however, explicitly mention aging or aging management. They were included in the review because they do address a number of the aging issues for various components.

Some examples of the ANS standards that can be related to implied aging are as follows:

- ANS-3.2 provides requirements and recommendations for administrative control, including written procedures for activities associated with NPP operation to help ensure that operations are carried out without undue risk to health and safety of the public. Among the activities are aging activities of inspection, testing and maintenance and repair.

- ANS-58.11 provides design criteria for achieving and maintaining cold shutdown conditions from a hot standby or post accident condition. This standard is explicit to safety; however, aging management of pressure and temperature cycles are addressed and adverse conditions are modulated by the suggested design criteria that confront nuclear safety issues of reactivity control, RCS integrity, P/T control, heat removal, and inventory control.

### 3.2.9 General Correlation of Implied Aging

While the above examples for all of the regulatory instruments and indeed the instrument review itself tend to correlate with the "implied" aging scenario, most of the instruments were developed to emphasize plant design and construction, plant pre-operational stages, and SARs. The instruments include concepts related to pertinent aging issues, but they were not written to solve aging problems or manage those issues. However, it is also true that some of the instruments apply throughout the life of the plant. The SARs are effective for the life of the plant and receive annual updates. Codes and standards, such as IEEE standards, that are noted in the SARs or other correspondence/agreements become a life-of-plant commitment. The Code of Federal Regulations apply throughout the life of the plant. The TSs apply throughout the length of the license and the ASME B&PV Code, Section XI, applies as long as the plant operates. In addition, the NRC Regulatory Guides apply and may become part of the license commitment if the utility commits to the guidance of the RG. GSI resolutions can be implemented throughout the life of the NPPs and become part of the licensing base.

# 4.0 CONCLUSIONS AND RECOMMENDATIONS

The following conclusions and recommendations were derived from the work completed to date on the five components selected for the first part, Volume 1, of the regulatory instrument review.

## 4.1 CONCLUSIONS

The following are the central conclusions derived from the review:

- The instruments included in this review, with minor exceptions, do not explicitly use aging or life extension terminology. The instruments, by definition, are dedicated to safety and have the ultimate purpose to establish and maintain safe operation of NPPs. Yet aging management does exist in them because safety-related design, construction and operation are consistent with the principles needed to provide aging management and life extension. This is to say, the technology associated with providing safety is congruous with technology required for management of aging.
- The emphasis in the regulatory instruments is on design, construction, equipment qualification and on the final safety review that will provide a safe plant at start-up. Specific actions to address aging and aging management, on the other hand, are generally initiated after a plant is placed in service.
- Revisions should be made to the instruments that explicitly address aging, see 4.2, Recommendations. Revisions will be difficult to achieve. Revisions of an instrument by instrument basis is the most practical approach at this time.

## 4.2 RECOMMENDATIONS

The following are the primary recommendations identified as a result of the review:

- The content of the regulatory instruments need to be changed to explicitly address the consequences of component, systems or structures (CSS) aging rates. Aging (degradation) rates are often not explicitly addressed in the body of the instruments and a key principal in the management of aging is understanding the rate of degradation over time. Emphasis on CSS aging rates, in the body of regulatory instruments, will serve five purposes: 1) establish an explicit need in NPP functional design criteria; 2) accelerate the improvement in trending and the methods of obtaining trending data; 3) accelerate the upgrading of applicable codes and standards; 4) accelerate the improvement in NDE tools and methods; and 5) improve CSS design and improve the design margins or the factors of safety needed.

- Project planning for revisions to the instruments will be a key feature in the success of any revision strategy and should address, as a minimum, the following questions:

    1. Should the individual instruments be changed or should a new comprehensive regulatory guide be prepared to address the aging management issue?

    2. What is the status of current industrial aging-related activities, such as EPRI research efforts, Nuclear Management and Resource Council (NUMARC) and Nuclear Plant Life Extension Committee (NUPLEX) reports, industry sponsored working group guidelines, documents and standards? (All sound solutions to aging problems should be included in the revision process.)

    3. What is the status of existing national society codes and standards relative to <u>aging initiatives</u>? This status check should include codes and/or standards of societies such as ASME, ANS, ANSI, National Fire Code (NFC), American Society of Testing and Materials (ASTM), American Concrete Institute (ACI), IEEE, and PLEX. (ASME has been given the overall PLEX responsibility for all standards groups.)

    4. What is a realistic time frame for implementation of the revisions?

# 5.0 SUGGESTED FUTURE REGULATORY INSTRUMENT REVIEW ACTIVITIES

The following are activities that should be considered for future work associated with this review task.

## 5.1 CONTINUATION OF THE GUIDE TO REGULATORY INSTRUMENTS

During the course of this instrument review, a Technical Evaluation Report containing the results of the review on LWR reactor pressure vessels was published: PNL-6848, Guide to Regulatory Instruments for LWR Reactor Pressure Vessels: Aging and License Renewal Considerations. The results contained in PNL-6848 are included in this report as Appendix II. It is suggested that the results of all the reviews, Appendix III, IV, and V and the reviews of FY 1990, be added to PNL-6848.

## 5.2 COMPUTER DATA BASE FOR THE REGULATORY INSTRUMENT REVIEW

It is suggested that the basic data from the Regulatory Instrument Review, Appendixes II through VI, be installed on a computer data-base system to make the information available for broader use. Information collected on other components should be added to the data base as it becomes available.

## 6.0 OBSERVATIONS OUTSIDE THE SCOPE OF THE REVIEW

The following are observations and recommendations that arose out of the review results that are not directly related to the objectives of the instrument review. They are based on the analysis of the instruments used to conduct the review, mainly on a close review of the comments included in the tables in Appendixes II through VI. For final validation, these observations should be subject to further study.

### COMPONENT REPLACEMENT

In general the regulatory instruments reviewed do not address component replacement as an aging management tool. Many components have design features for replacement and have been replaced well within the 40-year license period. Components such as, pumps, valves, motors, electrical and instrument controls fall within this category and are routinely replaced. Some components, notably the SG and primary piping, were never intended to be replaced; yet, a number of these replacements have been made within the 40-year license period. Some replacement direction is provided through components under TS surveillance and replacements will be made if repairs cannot be successfully performed. Also, electrical qualified (EQ) components within containment will be replaced according to their lifetime qualification testing. Thus, the scenario is that replacements are taking place because of design and necessity and within the jurisdiction of regulations/codes. The emphasis of the body of instruments, however, does not address replacement as a management tool. It is concluded that the instruments revisions should emphasize replacement methods by encouraging design for routine replacement and by encouraging improvement in CSS life assessment methods, including improved material degradation trends and component life predictions.

A factor in replacement methods is Diminishing Manufacturing Sources (DMS) or the increased unavailability of parts and components from manufacturers. DMS results in an increased vertical unsupportability of aging components and subsystems. DMS also results in an increase in maintenance costs. The NRC has identified DMS problems (NRC 1986) within the nuclear industry. These problems have resulted in down time and the necessity to operate in a degraded mode because of unavailable replacement parts. DMS has been identified by the Department of Defense (DOD) as a serious and prevalent problem. The DOD has, however, taken aggressive action to counteract the DMS phenomenon, e.g., using performance specifications rather than plant specifications, using improved long range forecasting and using improved procurement techniques. It is suggested that improved component NPP replacement methods should recognize the DMS problem and actively pursue solutions similar to the DOD strategy.

### IMPROVED ISI METHODS AND TOOLS

The regulatory instrument review revealed that the management of aging could be enhanced with improved ISI tools, e.g., Appendix II, page II.13, item (83) of this review for the reactor pressure vessels, with reference to

RG 1.150, states that improved flaw detection for the distribution of microflaws is needed. Also Appendix III, page III.13, item 92, addresses the aging issue, embrittlement, for the steam generator, with reference to 10 CFR 50, Appendix A, Criterion 51. Criterion 51 stipulates that the containment pressure boundary be designed for operating, testing and postulated accidents such that the ferritic materials behave in a non brittle manner. The Aging Needs, Column 7 for item 92 states that "Methods to measure embrittlement properties" are needed. The Life Extension Needs Column 8 states that "Improved NDE techniques to determine embrittlement properties" are needed. Therefore, for item 92, the review recognizes the need for improved NDE to provide in-situ embrittlement evaluation to verify that the containment material is behaving in a non brittle manner as stipulated in Criterion 51. Based on the results of the review, research is needed to improve traditional NDE and develop new NDE techniques for management of aging in the following areas:[a]

1. Effective means to determine fracture toughness and strength, and material embrittlement, including embrittlement induced by irradiation, hydrogen, and thermal changes. For example, embrittlement appears as a problem in nine out of eleven priority rankings (NRC 1987b) for PWR components shown in the priority rankings of NPAR degradation mechanisms.

2. Assessment of magnetic methods for material properties measurements to detect aging degradation of NPP steel materials should be encouraged and continued. Magnetic methods (Jiles 1988) may provide the techniques and have advantages over other NDE for in-situ evaluations especially as applied to the prediction of fatigue or creep damage.

3. Assessment of the use of acoustic emission (AE) for evaluating the continued serviceability of materials should be encouraged. AE is a unique tool that should have capabilities in environmental hardening, toughness, fatigue, and yield strength (Spanner 1979; Spanner 1985; Dal Re 1986).

4. Additional improvements are needed for the evaluation of stress corrosion cracking and intergranular attack on the outside diameter and inside diameter of steam generator tubes. Current technology has problems in this area; in most cases, cracks in the U bends or tube sheet area cannot be detected until they are through-wall. NDE technology improvements are also needed in vessel evaluation. Current methods cannot, with proven accuracy, detect shallow flaws of most interest in vessel fracture evaluations.

5. Studies are needed to determine NDE time intervals for inspection and the amount (coverage) of inspection needed to manage the aging process. (ASME Risk-Based and Reliability studies should provide a probabilitic basis for optimizing the level of NDE, as well as the intervals.)

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(a) This observation supports the need for the current work in NDE to improve existing techniques and develop new ones.

## WATER CHEMISTRY CONTROL

Water chemistry control is an important aging factor in the operation of LWRs, including steam generator tube corrosion in PWRs. Although considerable work on this factor has been conducted, especially in the 1970s, more research may be required before existing guidelines and/or instruments can be revised. As noted in the review, Appendix III, page III.17, item 125, control of primary side water chemistry is technically controlled by the TSs. The comments of item 125 (Column 9) note that secondary side water chemistry is administratively controlled and not controlled by the TSs. The comments indicate that improvements should be made in the management of the SG's secondary water side chemistry.

## EMPHASIS ON THE ASME CODE

The ASME BPVC Code is a principal instrument used in the design and construction of LWRs and it provides significant guidance and insight into the management of aging. The code is a living document with ongoing revisions, and many of the aging issues noted in this review are currently being addressed by ASME code committees and special working groups, e.g., the current PVRC Section III/XI committee, is addressing code limitations including PLEX. However, special emphasis needs to be placed in the following areas:

1. Code revisions are needed that more vigorously address material and/or component deterioration as functional criteria of the design.

2. The Code's emphasis on plant design requirements that allow for component ISI, repairs and replacements should continue. Improvements in the code that prescribe alternate or better design methods for ISI, repairs and replacements should be encouraged.

3. It has been estimated that a change in the Code takes about eight to ten years to fully implement. When appropriate, shorter time intervals are needed for implementing changes in the Code. The time cycle for specific items that did not appear in earlier codes needs to be shortened. (In some instances the time cycle has been shorten; it can be done. For example a new subsection for Section XI, IWT on erosion and corrosion, was approved by ASME's Main Committee within a three year time frame and in two years after ASME's Council on Codes and Standards gave the go-ahead. An additional problem, however, exists after the ASME involvement and that is the implementation through 10 CFR 50, 50.55a Codes and Standards. Thus, a change in code can also be coupled with the implementation delays. ASME Code Case(s) procedures and the RGs that identify acceptable cases certainly help the overall picture, but they are not the total answer.)

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# APPENDIX I

## GLOSSARY AGING ISSUES

## APPENDIX I

## GLOSSARY OF AGING ISSUES

**corrosion** - Corrosion is an aging issue that applies to all components in varying degrees and is manifested in different ways. Corrosion in the broad sense is the deterioration of material surfaces, chiefly, through electrochemical reactions and is influenced by the environment of the material or component, e.g., high humidity. A familiar example is the rusting of iron. Corrosion is also linked to other aging issues by causing an acceleration of the degradation, e.g., stress corrosion cracking, irradiated assisted corrosion cracking, corrosion fatigue, and corrosion/erosion.

**corrosion/erosion** - Corrosion/erosion occurs when the eroding fluid (liquid or gas) or particulate matter is in the presence of or contains corrosion causing products, i.e., the corrosion reaction is accelerated by the relative movement of the corrosive fluid and the metal surface. This mechanism involves the effects of mechanical wear or abrasion. Characteristics of corrosion/erosion are the appearance of grooves, gullies, waves, round holes and valleys on the surface of the component. An example of corrosion/erosion is tube wall thinning caused by impingement of water droplets containing suspended solids.

**crack formation and flaw growth** - Material crack formation and fatigue growth of flaws are indicators of material degradation and possible failure by through-wall cracks. A crack is a discontinuity at a particular location in a material as a result of localized excessive elastic/plastic deformation. A crack will propagate as long as the tensile stress acting on the component supplies sufficient energy to sustain a new crack surface (this aging issue is related fatigue and SCC, i.e, without a driver a crack will not grow). Present methods of flaw and crack detection are not 100% effective and improvements are needed for evaluation of crack initiation and growth. The issue also encompasses human factors of NDE staff, e.g., training.

**creep/swelling** - Creep is defined as the progressive deformation of a material at constant stress. Creep failure (fracture) or stress rupture exhibits the influence of relative high temperatures on long-time load-bearing characteristics. The creep temperature is often expressed as a homologous temperature, i.e., the ratio of the operating temperature T to the absolute melting temperature $T_m$ (K). Creep of load carrying components becomes of importance at a homologous temperature greater than a ratio of 0.50. Load carrying components operating in the temperature range of 538°C to 870°C (1000°F to 1600°F) are susceptible to creep. Generally, creep is not a major problem in LWRs because of their temperature operating levels. These levels at around 550°F are below the creep range for ASME Code materials specified in ASME BPVC, Section III. This aging mechanism, however, is of interest to the NPAR program because of its impact on electrical components/systems. An increase in the dimensions of components (e.g., fuel elements, and fuel cladding) swelling is caused by the accumulation of fission product atoms in the structural lattice of components. The fission products occupy a larger volume than the original material which caused swelling.

**element burnout(s)** - Element burnouts and repair and replacement of heaters are identified as an aging issue for the pressurizer vessels.

**erosion** - Erosion is a broad aging issue that is principally associated with fluid flow in components. High velocity water impingement can erode the walls of pipe and fittings. By definition pure erosion is the mechanical or wear action of a fluid and/or particulate matter on the surface of component parts. Erosion is usually accelerated by the presence of solid particles in flowing fluid. The possible consequences of typical erosion is the thinning of a pipe wall to failure.

**examination, inspection and maintenance and repair** - These activities may be considered aging issues because the management of aging is affected by these activities and in some cases aging is accelerated by poor examination, harsh inspection, and improper maintenance and repair. Also conditions may exist that do not permit adequate performance of the activities.

**excessive/harsh testing** - Testing programs and procedures that require excessive testing can lead to accelerated aging, e.g., harsh testing programs associated with the emergency diesel generators have been identified as contributors to aging.

**fatigue** - Fatigue in reactor operations is broadly defined as structural weakness and loss of resiliency in materials resulting from stressors, e.g., vibrations and thermally-induced fatigue cracks. Fatigue is defined as the phenomenon leading to fracture under repeated or fluctuating stresses having a maximum value less than the tensile strength of the material. Stress induced by stratified flow is an example of thermal fatigue. As the name implies, in stratified flows the cool water flows on the bottom of the pipe while steam flows on the top. These conditions can cause temperature differences that can lead to thermal cycling and fatigue of piping systems, e.g., pressurizer spray line and vessel nozzles.

**fracture toughness** - Fracture toughness is a material property that relates to the ability to resist fracture. Fracture toughness is related to the unique stress intensity level that causes failure in a component (is also called the critical stress intensity level). Its measure is the stress intensity factor, which incorporates both geometrical terms (the crack length appears explicitly, while the crack tip radius is assumed to be very sharp) and the stress level. It is an embrittlement aging issue specific to NPPs because ferritic materials suffer from decrease in fracture toughness as a result of irradiation. The CFRs specifically provide limits for the RPV material tests, i.e., initial Charpy tests of material shall be 75 ft-lb (102J) and throughout the life of the vessel shall not be less than 50 ft-lb (68J). Additionally, the material $RT_{NDT}$ is limited by the PTS screening rule.

**intergranular stress corrosion cracking (IGSCC)** - IGSCC is a preferential corrosion at the grain boundaries of a susceptible metal or alloy in the presence of a chemically aggressive environment, e.g., hot oxygenated water, and a tensile stress. IGSCC has occurred in austenitic SS in oxidizing environments such as BWRs. It also has occurred in SG tubes, safe ends, and type 304 SS piping. The principal concern is that SCC can cause ruptures, leakages, and plant shutdowns.

**irradiation embrittlement** - Irradiation embrittlement is defined as a decrease in fracture toughness due to long-term exposure to nuclear radiation. High neutron fluence levels can cause embrittlement in the RPV beltline region as well as other reactor internals and core supports. A reduction in tensile ductility is also caused by the neutron exposure.

**low-flux long-time irradiation of vessel, vessel internals and supports** - This aging issue is principally identified for the RPVs and vessel supports. Long-term flux reduces toughness and initial margins of safety in RPVs.

**seismic failure/damage** - Aging issues associated with earthquake damage were considered. Although not a classic aging issue, NPPs are required to evaluate seismic conditions and the consequences of natural phenomena and the damage that may result from these events. Aging could weaken a component so that it would not be able to withstand a seismic event.

**specific environmental factors, e.g., moisture, oxidation, chemicals, oils, and dust** - Environmental factors can induce aging in many NPP components. For this review the factors of moisture, chemicals, oils, dust, etc., are primarily of concern in the operation, testing, and performance of the EDGs. Because these factors can induce aging, physical location, surroundings, housekeeping and maintenance are important to EDGs. All these factors or elements can accelerate many of the aging mechanisms associated with the operation of diesel engines and their support equipment.

**stress corrosion cracking (SCC)** - SCC is degradation associated cracking accelerated by the combined effect of constant tensile stress, corrosion environments and susceptible microstructures. The stress may be residual in the material, as from cold working or heat treatment, or it may be externally applied. The observed crack may be transgranular or intergranular, depending on the nature of material and the environment. This term is a broader aging term than IGSCC and has been linked to bolting degradation in many cases as a principal aging problem, e.g., leaking steam through gaskets can react with the bolting lubricants and cause SCC.

**stud failure** - This aging issue is a factor in degradation of closures, flanges, manways, etc., through the cracking of bolts and studs from stress corrosion, fatigue, and corrosion.

**thermal cycles** - Aging issues are aligned with temperature cycles. Thermal cycling induces stress through thermal gradients/temperature changes. Thermal cycling induces thermal stress, low-cycle thermal fatigue, and high cycle fatigue (water temperature fluctuations). Low-cycle fatigue is defined as fatigue caused by high stresses and low numbers of cycles. Significant plastic strains occur during each cycle. Cycle lives are less than $10^4$ to $10^6$. High-cycle fatigue is defined as fatigue caused by low stresses and high numbers of cycles. Strain cycles are in the elastic range. Cycle lives are greater than $10^4$ to $10^6$.

**thermal aging embrittlement** - Thermal embrittlement is defined as a reduction in the ductility of a susceptible material due to a chemical change influenced by high temperature for long times. Thermal embrittlement of cast stainless steel components (pump housing, valve bodies, piping and fittings, etc.) is possible over periods of many years, resulting in increased critical flaw sizes. Thermal aging can significantly reduce fracture toughness and ductility of LWR reactor components materials. The rate of thermal embrittlement generally increases with increase in temperatures; however, specific material compositions of various steels are factors in thermal embrittlement.

**thermally induced bending** - This aging issue is principally (for this review) associated with the pressurizer caused by high-and low-cycle thermal loads at the water-steam interface in the vessel wall. (Thermally induced bending stresses can be a factor at other locations).

**thermally induced mechanical wear** - This aging issue is principally (for this review) associated with the heater failure in pressurizers caused by wearing and thinning of the heater due to rubbing action with supports because of thermal growth.

**transient thermal and pressure loads** - Transient or short thermal loads have been identified (for this review) as the transients effecting the pressurizer during heatups, cooldowns, testing, and abnormal events. For example, transients can occur in a few seconds and cause temperature changes of 55°C (100°F) in the surge-line nozzle.

**tube specific - denting, fretting and crevices** - These aging issues are SG tube specific. Denting results from crevice corrosion of the tube support plate and tubesheet materials; corrosion of carbon steel support plate and tubesheet squeezes the tube outside diameter and can result in decrease of thermal efficiency and cracking in the tube. Crevices provide areas for localized corrosion due to stagnant solutions in lap joints, holes, welding surfaces etc. Fretting in SGs results principally from wear action between the tube and tube supports and antivibration bars.

**vibration** - Vibration is a broad aging issue that can cause degradation in many of the components. Vibration by definition is any physical process which produces cyclic variations or motion. It is an aging issue to the extent that the vibration may cause failure or damage to a component. Excessive vibration can damage components by wear and promote material mechanical fatigue and pipe weld fatigue cracking. Vibration is a dynamic action and is associated in this review primarily with piping systems and the EDG. Dynamic loads are caused by response of the component to an oscillating input, e.g., flowing water or a rotating equipment.

**wear** - Wear usually results from the relative motion between two surfaces. Wear is the removal of discrete particles of material from a solid surface by the relative motion between that surface and one or more contacting surfaces. Wear is accelerated by hard and abrasive particles or action of fluids (erosion), vibration, or corrosive environments. The common result of wear is the reduction of thickness or damage to the mating surfaces. Wear is of a concern in rotating machinery, e.g., in pumps and EDGs and the other equipment/parts, e.g., valves stems and seats, and other mechanism subject to motion and vibration.

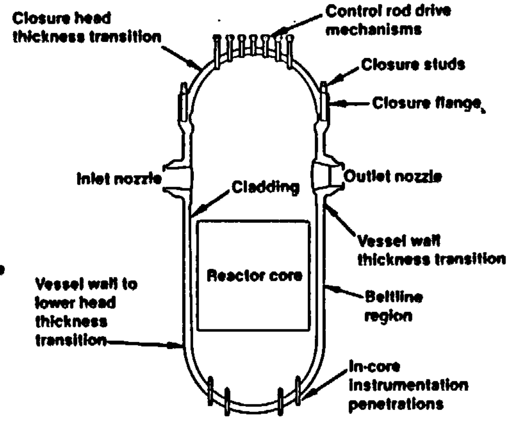
**weldments - specific to dissimilar metals of safe ends** - Weldments are formed by the welding together an assembly of pieces such as pipe to nozzles. Safe-end weldments have specific problems due to dissimilar metals. Among these are low-cycle thermal and mechanical fatigue, and IGSCC in the heat affected zone (HAZ) in the base metal.

# APPENDIX II

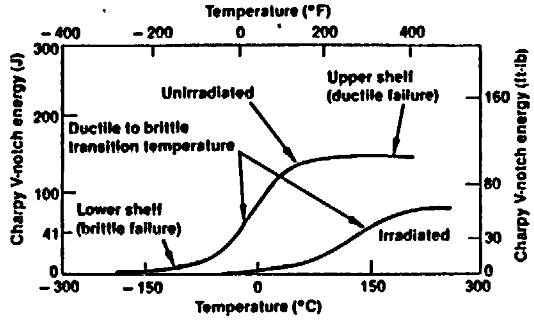
## <u>REGULATORY INSTRUMENT REVIEW FOR REACTOR PRESSURE VESSEL</u>

# Understanding and managing aging of PWR reactor pressure vessels

| | | |
|---|---|---|
| **Materials** | Vessel | · Low alloy carbon steel - SA-533B-1, SA-508-2, SA-302B |
| | Cladding | · Type 308 SS and 309 SS, |
| | Weldments | · Submerged arc (granular flux - linde 80, 91, 124 and 1092 manganese-molybdenum nickel filler wire) narrow gap submerged arc, shielded metal arc, and electroslag |
| | Closure studs | · SA-540 Gr. B24 Class 3 |
| **Stressors and Environment** | Neutron flux and fluence, temperature, reactor coolant, cyclic thermal and mechanical loads, preloads, and boric acid leakage | |




Typical PWR vessel showing important degradation sites.

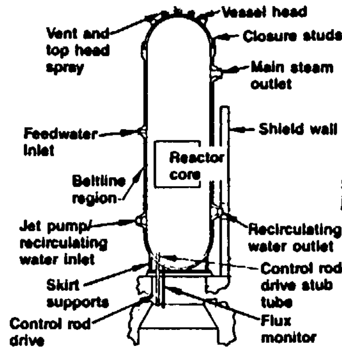



Effect of irradiation on the Charpy impact energy for a nuclear pressure vessel steel.

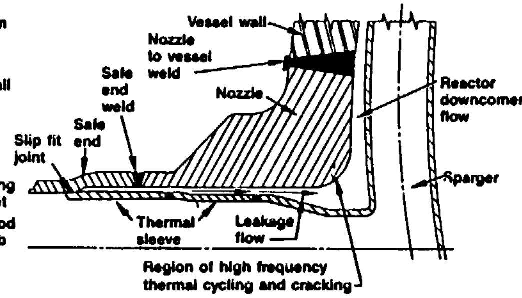
| UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions) | | MANAGING AGING | | |
|---|---|---|---|---|
| **Sites** | **Aging Concerns** | **Inservice Inspections, Surveillance, and Monitoring** | | **Mitigation** |
| | | **NRC Requirements** | **Recommendations** | |
| Beltline region | Irradiation embrittlement<br>- Chemical composition of vessel materials (Cu, Ni, P)<br>- Drop in upper shelf energy (USE)<br>- Shift in reference nil-ductility-transition-temperature ($RT_{NDT}$)<br>Environmental fatigue | Surveillance program to assess irradiation damage, i.e., shift in $RT_{NDT}$ and drop in USE (10 CFR 50 App. H, Reg. Guide 1.99, Rev. 2)<br>Pressurized thermal shock (PTS) screening criteria (10 CFR 50.61)<br>PTS rule, RG 1.154<br>Damage evaluation (10 CFR 50 App. G)<br>Pressure - Temperature (P-T) limits during heatup, cooldown, criticality, and inservice leakage and hydrostatic pressure test to prevent nonductile fracture (Tech. Spec. requirement, 10 CFR 50 App. G)<br>[P-T limits are also applied to non-beltline region]<br>Low temperature overpressurization (LTOP) protection setpoint (Tech. Spec. requirement)<br>Volumetric examination of all welds during each inspection interval (10 CFR 50.55a, IWB-2500, Reg. Guide 1.150, Rev. 1)<br>Flaw evaluation (10 CFR 50.55a, IWB-3000)<br>Leakage and hydrostatic pressure tests (10 CFR 50.55a, IWA-5000) | Include fracture toughness and tensile test specimens in surveillance program<br>Develop use of reconstituted and miniature specimens<br>Develop techniques for in situ determination of mechanical properties<br>Perform accelerated irradiation tests of reconstituted specimens<br>Revise Reg. Guide 1.99, Rev. 2 to account for phosphorus with low copper<br>Use state-of-the-art ultrasonic inspection techniques for improved reliability of defect detection, sizing, and characterization<br>- Automated amplitude-based systems<br>- Tip diffraction techniques<br>- Large-diameter focused transducer<br>Use fatigue crack growth curves (ASME SC XI, Appendix A)<br>Develop acoustic emission monitoring to detect crack growth (Nonmandatory appendix is being developed for ASME Section XI) | Neutron flux reduction<br>Inservice annealing (ASTM E 509-86)<br>Determine effects of annealing and reembrittlement rate |
| Outlet/inlet nozzles | Environmental fatigue<br>Irradiation embrittlement<br>Function of nozzle elevation<br>(Potential impact of Reg. Guide 1.99, Rev. 2) | Volumetric examination of all nozzle-to-vessel welds and nozzle inside radius sections during each inspection interval (IWB-2500)<br>Volumetric and surface examination of all dissimilar metal welds during each inspection interval (IWB-2500) | Use on-line fatigue monitoring (monitoring of pipe wall temperatures and coolant flows, temperatures, and pressures)<br>Evaluate irradiation embrittlement damage | |
| Instrumentation nozzles<br>CRDM housing nozzles | Environmental fatigue | Visual examination of external weld surface of 25% of nozzles during system hydrostatic test (IWB-2500) | | |
| Closure studs | Environmental fatigue<br>- preload cycles during head replacement<br>Boric acid corrosion (if leakage occurs) | Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval (IWB-2500) | | |

# Understanding and managing aging of BWR pressure vessels

| | | |
|---|---|---|
| Materials | Vessel | - Low alloy carbon steel SA-533B-1, SA-302B |
| | Cladding | - Type 308 SS or 309 SS, |
| | Nozzles | - SA-508-2 |
| | Safe Ends | - Type 304 SS, Type 316 SS, Inconel SB-166, Inconel SB-167, SA-508-1 |
| | Thermal Sleeves | - Type 304 SS |
| | Closure Studs | - SA-540 Gr. B22 or B23 |
| | Weldments | - SA-193 Gr. B7 |
| Stressors and Environment | Operational transients, neutron flux and fluence, temperature, and reactor coolant | |



Typical BWR pressure vessel



Cross section of feedwater nozzle with cracking location

| UNDERSTANDING AGING (Materials, Stressors, and Environmental Interactions) | | MANAGING AGING | | |
|---|---|---|---|---|
| **Sites** | **Aging Concerns** | **Inservice Inspection, Surveillance, and Monitoring** | | **Mitigation** |
| | | NRC Requirements | Recommendations | |
| Feedwater nozzles and safe end welds | High-cycle thermal fatigue caused by feedwater leakage<br>Environmental fatigue | Volumetric examination of all nozzle-to-vessel welds and nozzle inside radius sections during each inspection interval (IWB-2500)<br>Volumetric and surface examination of all dissimilar metal welds during each inspection interval (IWB-2500) | Use on-line fatigue monitoring (monitoring of pipe wall temperatures and coolant flows, temperatures, and pressures)<br>Develop criteria for assessing high-cycle fatigue damage | Modify design, revise operating procedures, and remove feedwater nozzle cladding to prevent fatigue cracking |
| Recirculation inlet/outlet nozzles and dissimilar metal welds | IGSCC crack initiated in HAZ may propagate into base metal<br>Environmental fatigue | | Develop on-line corrosion monitoring<br>Evaluate long-term effects of hydrogen water chemistry<br>Develop robotics system for remote inspection probe positioning and scanning | Implement hydrogen water chemistry to reduce IGSCC damage |
| Welds<br>- Control rod drive stub tubes<br>- Interior attachments | IGSCC crack initiated in HAZ may propagate into base metal by corrosion and/or environmental fatigue | Visual examination of all accessible interior attachment welds during each inspection interval (IWB-2500) | | |
| Beltline Region | Irradiation embrittlement<br>- Chemical composition of vessel materials (Cu, Ni, P)<br>- Drop in upper shelf energy (USE)<br>- Shift in reference nil-ductility-transition-temperature ($RT_{NDT}$)<br>- Welds are more susceptible than base metal<br>- Flux is lower than that in PWR vessel<br>Environmental fatigue | Surveillance program to assess shift in $RT_{NDT}$ and drop in USE (10 CFR 50 App. H, Reg. Guide 1.99, Rev. 2)<br>Damage evaluation (10 CFR 50 App. G)<br>Pressure-temperature (P-T) limits during heatup, cooldown, criticality, and inservice leakage and hydrostatic pressure tests to prevent nonductile fracture (Tech. spec. requirement, 10 CFR 50 App. G.)<br>[P-T limits are also applied to non-beltline region]<br>Volumetric examination of all shell welds during each inspection interval (10 CFR 50.55a, IWB-2500, Reg. Guide 1.150, Rev. 1)<br>Flaw evaluation (10 CFR 50.55a, IWB-3000)<br>Leakage and hydrostatic pressure tests (10 CFR 50.55a, IWA-5000, IWB-5000) | Revise Reg. Guide 1.99, Rev. 2 to account for phosphorous when copper content is low<br>Use state-of-the-art inspection techniques for improved reliability of defect detection, sizing, and characterization<br>Develop robotics system for remote inspection probe positioning and scanning<br>Include fracture toughness and tensile test specimens in surveillance program<br>Develop use of reconstituted and miniature specimens and accelerated irradiation of reconstitued specimens<br>Use fatigue crack growth curves (ASME Section XI, Appendix A)<br>Develop acoustic emission monitoring to detect crack growth (nonmandatory appendix is being developed by ASME Section XI) | Inservice annealing (ASTM E 509-86)<br>Determine effects of annealing and reembrittlement rate<br>Implement neutron flux reduction program |
| Closure Studs | Fatigue, fretting | Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval (IWB-2500) | | |
| External attachment welds such as skirt supports | Low-cycle thermal and mechanical fatigue | Volumetric or surface examination (IWB-2500) | | |

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## REGULATORY INSTRUMENT REVIEW FOR REACTOR PRESSURE VESSEL

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| RPV (1) | EXAMINATION, INSPECTION AND SURVEILLANCE (EIS) | 10 CFR 50 Appx. A Criterion 32 | The RPV should be designed to permit inspections. | NONE | Appropriate measures are being defined for plants where inspections are difficult. | N/A: Design doesn't change over time. | RPVs need to be considered for acceptability if critical areas are not easily inspected. | The pressure boundary should be designed to permit inspections and appropriate material surveillance programs for the RPV. New aging related criteria may need to be written and referenced by this criterion. Currently the plants are required to apply criterion 32 or make a best effort to inspect the RPV. |
| (2) | | 10 CFR 50, Appx. G (III) | All inspection programs as per the ASME codes. | NONE | NONE | NONE | Inspections may need to use different procedures aimed at detecting aging/degradation for life evaluation including irradiation and fatigue failure modes. | Should reflect life extension decisions and aging research. Evidence for aging of RPV hardware includes erosive degradation of thimble tubes. A need exists for a new base line examination. |
| (3) | | 10 CFR 50<br><br>Appx. G (V) - B.<br><br>C. | Surveillance and/or inservice examination for fracture toughness<br><br>Alternate criteria involving ISI per ASME Section XI | NONE | NONE | Evaluate effects of age on the material properties used in the calculations (see embrittlement issue). | Are new operating limits and inspection frequencies needed for life extension? Inspection criteria for irradiation and fatigue failure modes may be needed. | 10 CFR 50 Appx G (IV) requirements must be satisfied prior to life extension. Requirements for operation is 10 CFR 50 Appx. G (IV) is not satisfied: Beltline flaws examined as per ASME Section XI. Work is needed to include life extension decisions and aging research. Inspection exemptions will need to be reconsidered for life extension. |
| (4) | | 10 CFR 50, Appx. H | Capsule survey program. | NONE | Known to exist. | Assessment of the withdrawal schedule and number of samples may be needed. Alternate testing methods (i.e. mini-samples) should be considered. | Assessment of the withdrawal schedule and number of samples may be needed. Capsules may need to be reinserted or pre-irradiated capsules may need to be inserted. Annealing effects need investigation. | No material surveillance program is required for RPVs for which it can be demonstrated the peak neutron fluence at the end of the design life will not exceed $10^{(17)}$ n/cm$^2$ (E >1 Mev). The capsule program should be as per ASTM E-185. Capsule locations and withdrawal schedules are required. Modifications to ASTM E-185 for capsule withdrawal scheme for new fuel management programs are in progress. Similar designed and operated reactors may use integrated surveillance programs. |
| (5) | | Tech Spec 4.0.5 | References ASME Section XI. | Repair. | NONE | Advances in available NDE technology are needed to detect shallow flows of most interest in vessel fracture evaluations. | The unreliability of detection/sizing flaws because of cladding effects should be addressed. Human factor effects of NDE staff should also be addressed. | Current flaw detection systems are not 100% effective. Detection systems should be improved. Accessibility to BWR PRVs is a problem; remote scanners are needed. |
| (6) | | Tech Spec 4.4.9.1.2 | Capsule Survey Program | Surveillance of RPV transient temperatures | NONE | Testing sample frequency. Alternate testing methods (i.e. mini-samples) should be considered. | Assessment of accelerated irradiation effects. | Extend license if sufficient margins exist. |

General Notes:
1. A "?" indicates further study/investigation is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessarily the aging issue.
3. For meaning of abbreviations, acronyms, and initialisms, used throughout, see acronyms on page xi, xii, and xiii of the report.

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (7) | EIS (contd) | SRP 3.9.2.I.4 | NONE | NONE | NONE | N/A | Should the SRP address life extension? | Testing during pre-operational and startup test program; This section does not apply after the start of plant operations. |
| (8) | | SRP 5.2.1.2.II.2 | Inservice inspection code case and RG applicability. | NONE | NONE | Need to verify that the reference code cases cover an adequate time span for reactor operation. | NONE | Acceptable code cases for inservice inspection are found in Reg. Guide 1.147. Code cases must be reviewed every three years. |
| (9) | | SRP 5.2.4.II.4 | In service inspection intervals. | NONE | NONE | Does inspection frequency need to be increased in the last ten year interval? More extensive examinations may be needed. | New life extension document that reflects time greater than forty years? | Required inspections must be performed on reactor coolant pressure boundary components during each 10-year interval of service as per ASME Section XI IWA-2000. Option...impose inspection plan A. |
| (10) | | SRP 5.3.1.I.3 | Nondestructive examinations (NDE) | NONE | NONE | ? | Should the SRP address life extension? | Special methods for NDE other than those in ASME should be reviewed. |
| (11) | | SRP 5.3.1.I.6 | Surveillance data collection over vessel lifetime. | NONE | NONE | Assessment of the withdrawal schedule and number of samples may be needed. Alternate testing methods (i.e., mini-samples) should be considered. | Should the SRP address life extension? | RPV surveillance must be performed to monitor changes in fracture toughness properties. |
| (12) | | SRP 5.3.1.II.6 | Material Surveillance. | NONE | NONE | Assessment of the withdrawal schedule and number of samples may be needed. Alternate testing methods (i.e. mini-samples) should be considered. | Should the SRP address life extension? | No material surveillance program is required if it is shown that the fluence will be less than $10^{17}$ $n/cm^2$ (E > 1 Mev). If this is not met, a surveillance program is needed as per ASTM E185. Irradiation damage is a less severe problem for BWRs than for PWRs. |
| (13) | | SRP 5.3.1.III.6 | Fluence calculation or surveillance. | NONE | See note 1. | Assessment of the withdrawal schedule and number of samples may be needed. Alternate testing methods (i.e. mini-samples) should be considered. | Should the SRP address life extension? | End-of-life fluence must be less than the maximum or surveillance must be as per ASTM E185. |
| (14) | | SRP 5.3.3 BTP 5-2A | Fracture toughness, pressure, temperature, and surveillance requirements. | NONE | NONE | Appropriate regulatory instruments must be modified for aging. Alternate testing methods (i.e. mini-samples) should be considered. Review of circumferential welds and the effects of streaming are also needed. | Should the SRP address life extension? | The BTP provides a summary of the requirements for fracture toughness, pressure, temperature, and surveillance requirements as stated in ASME codes and 10 CFR 50 |

-- Reactor Pressure Vessel

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (15) | | RG 1.150 | UT of RPV welds. | Inservice inspection ISI) | ? | Better flaw detection systems are needed for distributed micro-flaws. | Human factors of NDE should be evaluated, i.e., flaws incorrectly sized should be avoided. | NDE of welds for crack initiation and growth. New detection systems would constitute a major R&D effort. Is 100% UT required for BWRs, to assess under cladding cracks? Cracks may propagate from cladding into steel in the RPV. |
| (16) | EIS (contd) | RG 1.154 | In situ test for fatigue. | NONE | NONE | Needs regulatory position development. | Needs regulatory position development. | Applies to pre-startup vibration tests. The RG does not consider fluence effects. |
| (17) | | RG 1.34 | Trace element effects on electroslag welds; residual elements not considered. | NONE | NONE | Assess effects of residual elements. | Needs regulatory position development. | Applies to pre-startup vibration tests. The RG does not consider fluence effects. |
| (18) | | RG 1.65 | NDE/UT for crack, etc. in studs. | ISI | NONE | Revise RG 1.65 | Revise RG 1.65 | The document allows for the monitoring of studs for damage with time. |
| (19) | | RG 1.2 | In situ test for fatigue. | NONE | NONE | Needs regulatory position development. | Needs regulatory position development. | Applies to pre-startup vibration tests. The RG does not consider fluence effects. |
| (20) | | ASME Sec XI<br>IWA-2200<br>IWA-2300<br>IWA-2400 | Identifies the rules and requirements for inspection responsibilities, accessibility, examination methods, personnel qualifications frequency and records. Minimum basis for all requirements. | NONE | Sec XI under review for life extension applicability | Alternate testing methods (i.e. mini-samples) should be considered. | N/A | Applies to the pressure retaining components only. The only life extension application is the design, fabrication, installation and inspection of replacement components. Specific requirements are found in ASME Sec XI IWB-2500. |
| (21) | | GSI 29 | Cracking and degradation of bolts and studs from stress corrosion, fatigue, boric acid corrosion and erosion corrosion. | NONE | NONE | Does the inspection frequency or fraction inspected need to be increased for extended life? Do records need to be maintained and analyzed for trends of replaced studs and bolts? | Further analysis needed. | Bolting degradation or failure in nuclear power plants<br>• Priority - HIGH<br>• Proposed inspecting 10% of bolts each outage, extending an outage 1.5 days.<br>• This issue is broader than studs and bolts for the PRV. |
| (22) | | GSI 79 | Axial temperature gradients that could cause thermal stress in the flange area or in the studs. Cycling over the life of the plant may reduce fatigue margin. | NONE | NONE | Further analysis needed. | Further analysis needed. | Unanalyzed RPV thermal stress during natural convection cooldown<br>• Priority -- MEDIUM |
| (23) | | GSI 94 | Major overpressure of the RCS, in combination with the existence of flaws, could result in a brittle failure of the RPV. | NONE | NONE | Further analysis needed. | Further analysis needed | Additional low temperature overpressure protection for LWRs<br>• Priority -- HIGH |
| (24) | | GSI 111 | Stress corrosion cracking | NONE | NONE | Further analysis needed. | Further analysis needed. | Stress corrosion cracking of pressure boundary ferritic steels in selected environments<br>• Priority -- LOW<br>• Classified as Licensing Issue |

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| (25) | EIS (contd) | GSI A-11 | As plants accumulate increasing service time, neutron irradiation reduces the material fracture toughness and initial margins of safety. | NONE | NONE | Monitor and analyze trends of Nil-Ductility temperature | Further analysis needed. | Reactor vessel materials toughness<br>• Priority -- RESOLVED with issuance of NUREG-0744 and Generic Letter 82-66<br>• See A-49 for a related discussion.<br>Reactor Pressure Vessel |
| (26) | | GSI A-26 | Toughness at low temperature | NONE | NONE | Further analysis needed. | Further analysis needed. | Reactor Pressure Vessel Transient<br>• Priority -- RESOLVED with publication of NUREG-0224 and SRP 5.2<br>• Numerous reported pressure transients in PWRs where TS pressure and temperature limits were exceeded. Majority at solid water condition of startup or shutdown and relatively low reactor temperature. Less toughness at low temperature, therefore more susceptible to brittle fracture. |
| (27) | | USI A-49 | Neutron irradiation of the reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. | NONE | NONE | Monitor and analyze trends of Nil-Ductility temperature. | Further analysis is needed to determine what impact life extension will have on the brittleness of RPV materials. | Pressurized Thermal Shock (PTS)<br>• Resolution for aging will be affected by licensee actions that include annealing and fuel management practices to reduce flux at reactor pressure vessel. |
| (28) | | GSI B-6 | Through wall cracks and fatigue crack growth | NONE | NONE | Further analysis needed. | Further analysis needed. | Loads, load combinations, stress limits<br>• Priority -- HIGH<br>• See NUREG-2800 Sup1 |
| (29) | | GSI C-12 | Structural damage by vibrations of sufficient magnitude. Fretting due to deficient design and material selection for anti-vibration bars. Fatigue failure, particularly at nozzles where stresses are highest. | NONE | NONE | Further analysis needed. | Further analysis needed. | Primary system vibrations<br>• Priority -- RESOLVED |
| RPV (30) | IRRADIATION EMBRITTLEMENT, FRACTURE TOUGHNESS (IEFT) | 10 CFR 50.55a | System and component designs shall meet ASME requirements. | NONE | NONE | Reference documents need to reflect aging. | Related documents need to be reviewed for life extension. | ASME requirements must be met throughout the operating life. ASME Section XI is currently under review for LR applicability. |
| (31) | | 10 CFR 50.60 | Acceptable criteria for fracture prevention. | NONE | NONE | Reference documents need to reflect aging. | New document section may be needed to address life extension. | References 10 CFR 50 Appx. G and H. |
| (32) | | 10 CFR 50.61 | Fracture toughness requirements to protect against PTS | NONE | See note 1. | Impact of aging on increased risk for PTS-related event occurrence needs to be addressed. | PTS needs to be better understood for the plant conditions that would exist after life extension. | Provides fracture toughness requirements for protection against pressurized thermal shock. Calculation of $RT_{(NDT)}$ for the RPV and the overlay clad irradiation damage. Requirements may need to be changed to reflect potential new failure modes caused by irradiation damage. A need exists for the improvement of meaningful flaw population data and for identifying the influence this data has on probabilistic fracture analysis. |

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| (33) | IEFT (contd) | 10 CFR 50, Appx. A Criterion 31. | Design Margin specifications. | NONE | NONE | Are margins quantified as a function of exposure or are they based only on initial design? | Evaluation of ASME Section XI, Appx. A for margins for extended duty; comparison to refurbishment activities such as annealing. | The pressure boundary should be designed with sufficient margin to prevent brittle fracture and rapid propagation of fractures. The design reflects material properties, effects of irradiation, the size of flaws, and the stresses that are experienced. This section does not apply after the start of plant operations. |
| (34) | | 10 CFR 50, Appx. G | Fracture toughness requirements for ferritic steels referenced to ASME Section III codes. | NONE | NONE | Enhanced NDE of vessel may be needed. | Effect of annealing on fracture toughness and crack tip geometries (micro-damage modification by diffusional mechanisms during annealing). | Should reflect life extension decision and aging research. Assumes the flaws of ASME Section XI, Appendix A. |
| (35) | | 10 CFR 50, Appx. G (III) | Fracture toughness requirements as per the ASME codes. All inspection programs as per the ASME codes. | NONE | NONE | Inspection scope may need to be increased. | Inspections may need to use different procedures aimed at detecting aging/degradation for life extension including irradiation and fatigue failure modes. | Should reflect life extension decisions and aging research. Evidence for aging of RPV hardware includes erosive degradation of thimble tubes. |
| (36) | | 10 CFR 50, Appx. G (IV) | Fracture toughness requirements. | NONE | NONE | Review of frequency of assessments of RPV $K_{IC}$ location on uppershelf and uppershelf toughness reduction. | Need to determine effects of annealing. | Includes Charpy upper shelf energy restrictions, and temperature limitations for operations and tests. Assumes the flaws of ASME Section XI, Appendix A. Requirements may need to be changed to reflect potential new failure modes caused by irradiation damage, fatigue, erosion, etc. |
| (37) | | 10 CFR 50, Appx. G (V) (B) | Inservice examination requirements. | NONE | NONE | Evaluate effects of age on the materials properties used in the calculations. | Are new operating limits and inspection frequencies needed for life extension? Inspection criteria for irradiation and fatigue failure modes may be needed. | 10 CFR 50 Appx G (IV) requirements must be satisfied prior to life extension. Requirements for operation if 10 CFR 50 Appx. G (IV) is not satisfied: Beltline flaws examined as per ASME Section XI; evidence of fracture toughness from tests; existence of sufficient safety margins. Work is needed to include life extension decisions and aging research. Inspection exemptions will need to be reconsidered for life extension. |
| (38) | | 10 CFR 50, Appx. H | Capsule survey program. | NONE | ? | Assessment of the withdrawal schedule and number of samples may be needed. Alternate testing methods (i.e. minisamples) should be considered. | Assessment of the withdrawal schedule and number of samples may be needed. Annealing effects need investigation. | No material surveillance program is required for RPVs for which it can be demonstrated the peak neutron fluence at the end of the design life will not exceed $10^{17}$ $n/cm^2$ (E > 1 Mev). The capsule program should be as per ASTM E-185. Capsule locations and withdrawal schedules are required. Modifications to ASTM E-185 for capsule withdrawal scheme for new fuel management programs are in progress. Similar designed and operated reactors may use integrated surveillance programs. |
| (39) | | Tech Spec 3/4.4.9 | P/T Curve Adjustments | Annealing, replacement | NONE | Extend radiation damage curve to higher fluences. | Need to determine the effects of annealing. | Trend data to project when flux reductions are needed. |

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| (40) | IEFT (contd) | Tech Spec 4.4.9.1.2 | Capsule Survey Program | Fuel Management | NONE | Testing sample frequency. Alternate testing methods (i.e., mini-samples) should be considered. | Accelerated Irr. effects. | Extend license if sufficient margins exist. |
| (41) | | SRP 5.3.1.I.1 | Material irradiation resistance. | NONE | NONE | Need to assess if design data bases are sufficiently complete with respect to fluence. | Should the SRP address life extension? | Deals with the initial design choice of materials. Adequacy for use of materials is to be assessed on the basis of mechanical and physical properties, effects of irradiation, corrosion resistance, and fabricability. This section does not apply after the start of plant operations. |
| (42) | | SRP 5.3.1.I.3 | Nondestructive examinations. | NONE | NONE | Alternate testing methods (i.e., mini-samples) should be considered. | Should the SRP address life extension? | Special methods for NDE other than those in ASME should be reviewed. |
| (43) | | SRP 5.3.1.I.5 | Fracture toughness calculation. | NONE | NONE | Many of the vessels with low upper-shelf weld problems are ring forged vessels. Present computational methods are based on axis-symmetric (2-D) analysis. The effects of streaming are thought to be potentially significant and this may require a 3-D analysis model with appropriate streaming simulation. Calculational methods are standard regardless of material. | Minor revision of existing document is needed to reflect the awareness of material property degradations. A data base reference for aged, annealed materials is needed, or the expected results should be standardized. | Fracture toughness tests must be performed on all ferritic tests specimen. Fracture toughness is characterized by a reference temperature. |
| (44) | | SRP 5.3.1.I.6 | Data collection over vessel lifetime. | NONE | NONE | Record keeping. | Should the SRP address life extension? | RPV surveillance must be performed to monitor changes in fracture toughness properties. |
| (45) | | SRP 5.3.1.II.6 | Material Surveillance. | NONE | NONE | Assessment of the withdrawal schedule and number of samples may be needed. Alternate testing methods (i.e. mini-samples) should be considered. | Should the SRP address life extension? | No material surveillance program is required if it is shown that the fluence will be less than $10^{17}$ $n/cm^2$ (E > 1 Mev). If this is not met, a surveillance program is needed as per ASTM E185. |
| (46) | | SRP 5.3.1.III.6 | Fluence calculation or surveillance. | NONE | ? | Assessment of the withdrawal schedule and number of samples may be needed. Alternate testing methods (i.e. mini-samples) should be considered. | Should the SRP address life extension? | End-of-life fluence must be less than the maximum or surveillance must be as per ASTM E185. |

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| (47) | | SRP BTP-MTEB 5-2 A | Fracture toughness, pressure, temperature, and surveillance requirements. | NONE | NONE | Appropriate regulatory instruments must be modified for aging. Alternate testing methods (i.e. mini-samples) should be considered. | Should the SRP address life extension? | Summary of the requirements for fracture toughness, pressure, temperature, and surveillance requirements as stated in ASME and 10 CFR 50. |
| (48) | IEFT (contd) | RG 1.154 | In situ test for fatigue. | NONE | NONE | Needs regulatory position development. | Needs regulatory position development. | Applies to pre-startup vibration tests. The RG does not consider fluence effects. |
| (49) | | RG 1.99 Rev. 2 | Fluence/chemistry factors for transition temperature shift. | NONE | NONE | Need to account for phosphorous with low copper. | Irradiation series to fluences appropriate for life extension. Revision to RG for BWR P/T curves. | Conservative prediction of NDT shift as a function of fluence and alloy content. May constitute a penalty for BWR P/T curve restriction on RPVs. There is a revision of ASTM E-900 in progress. There is also a yet unnumbered RG for physics dosimetry. Irradiation effects and fracture toughness information for base metal welds and clads to high fluences needs to be evaluated. |
| (50) | | RG 1.XXX (New RG - not numbered yet) | Identifies the physics-dosimetry requirements for PV fluence calculations. Recommendations are based on current industry practice and state-of-the art dosimetry methods. | Licensee $RT_{PTS}$ Analyses | Under review for general applicability. | Appropriate development of any new standards. | Assessment of needs for any new standards. | This is a new regulatory guide. The methods and assumptions presented in this guide will provide an acceptable approach for determining pressure vessel damage (>1 Mev) fluences for input to the $RT_{PTS}$ prescription given in 10 CFR 50.61. |
| (51) | | ASME Sec III NB-2160 NB-2331 NB-3124 | NDT shift. | None, except for original design considerations. | Section under review for life extension applicability. | N/A | N/A | The code is based on fossil fuel experience and does not include fluence-caused problems. This section does not apply after the start of plant operations. |
| (52) | | ASME Sec III Appx. G-2000 | Protection against non-ductile vessel failure. | NONE | Section under review for life extension applicability | N/A | N/A | This appendix is non-mandatory. Contains procedures for obtaining allowable loadings for ferritic pressure retaining materials. Methods for calculating critical stress intensity factors, allowable pressures, and shell and head service limits are given. Nozzles, flanges and shell regions are considered. Service limits and hydrostatic test temperature limits are given. Requirements may need to be changed to reflect potential new failure modes caused by irradiation damage. $RT_{NDT}$ margins are being re-evaluated for RPV hydrotest. |
| (53) | | ASME Sec XI Appx. A. | Analysis methods for fracture toughness and irradiation effects. | NONE | Section under review for life extension applicability. There is a yet unnumbered RG for physics dosimetry that will relate to this section | N/A | N/A | This appendix is non-mandatory. It contains procedures for accepting flaws that exceed IWB-3500 standards. Includes rules and equations for fracture toughness determination, fatigue crack growth rate analysis, and calculation of irradiation effects on $RT_{NDT}$ for the RPV. It also includes crack growth rate curves. Requirements may need to be changed to reflect potential new failure modes caused by irradiation damage. |

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| (54) | IEFT (contd) | ASME Sec XI Appx. G | Fracture toughness criteria. See comments. | NONE | Section under review for life extension applicability. | A need exists to consider any special requirements for flaws in circumferential welds. | N/A | This appendix is non-mandatory. Fracture toughness criteria for protection against failure. For example, $RT_{NDT}$ margins are being re-evaluated for RPV hydrotest. Procedures for obtaining allowable loadings for ferritic pressure retaining materials. Calculation of stress intensity factors. Allowable pressure equations that can also be used for nozzles, flanges and shell regions. Service limits and hydrostatic test temperature requirements specified. Bolting requirements to prevent failure. |
| (55) | | GSI 94 | Major overpressure of the RCS, if combined with critical size crack, could result in a brittle failure of the reactor vessel. | NONE | NONE | Further analysis needed. | Further analysis needed. | Additional low temperature overpressure protection for LWRs<br>• Priority -- HIGH |
| (56) | | GSI A-11 | As plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial margins of safety. | NONE | NONE | Further analysis needed. | Further analysis needed. | Reactor vessel materials toughness<br>• Priority -- RESOLVED with issuance of NUREG-0744 and Generic Letter 82-66<br>• See A-49 for a related discussion. |
| (57) | | GSI A-26 | Toughness at low temperature | NONE | NONE | Further analysis needed. | Further analysis needed | Reactor Pressure Vessel Transient<br>• Priority -- RESOLVED with publication of NUREG-0224 and SRP 5.2<br>• Numerous reported pressure transients in PWRs where TS pressure and temperature limits were exceeded. Majority at solid water condition of startup or shutdown and relatively low reactor temperature. Less toughness at low temperature, therefore more susceptible to brittle fracture. |
| (58) | | USI A-49 | Neutron irradiation of the reactor pressure vessel weld and plate materials decreases fracture toughness of materials | NONE | NONE | Further analysis needed. | Further analysis needed. | Pressurized Thermal Shock<br>• For life extension analysis for resolution of GSI should address licensee actions such as annealing and fuel management. |
| (59) | | GSI B-6 | Through wall cracks and fatigue crack growth | NONE | NONE | Further analysis needed. | Further analysis needed. | Loads, load combinations, stress limits<br>• Priority -- HIGH<br>• See NUREG-2800 Sup1 |
| RPV (60) | FATIGUE | Tech Spec 4.4.9.1.2 | Cyclic loadings, P/T records. | Analysis, thermal barriers. | NONE | Actual loading analysis. | Evaluate cycle limits. | Reanalysis recommended to determine if cycle limits can be extended. Thermal barrier research needed. |
| (61) | | SRP 3.9.1 | Transient induced fatigue. Perform fatigue analysis in design stage. | NONE | NONE | N/A | Should the SRP address life extension? | SRP guides initial design. This section does not apply after plant startup. |
| (62) | | SRP 5.2.1.2.II.2 | Inservice inspection code case and RG applicability. | NONE | NONE | Adequate time for the use of code cases...Code cases expire or are reviewed every three years. | NONE | Acceptable code cases for inservice inspection are found in Reg. Guide 1.147. |

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| (63) | | RG 1.2 | In situ test for fatigue. | NONE | NONE | Needs regulatory position development. | Needs regulatory position development. | Applies to pre-startup vibration tests. The RG does not consider fluence effects. |
| (64) | | RG 1.48 | Prediction of cycles. | NONE | NONE | Needs revision; no enforcement capability. | Documentation of the actual service history needed. | Design loads are not the same as actual. Need to account A, B, C, D, loads throughout life. |
| (65) | Fatigue (contd) | ASME Sec III NB-3000 NB-3222.4 | Thermal and pressure cycles. Design and analysis. | None, except for original design. | Section under review for life extension applicability. | N/A | N/A | Plants now operating need to start counting and documenting the cyclic history of each plant so comparisons can be made to assumed design cycles following the design code. This section does not apply after the start of plant operations. |
| (66) | | GSI 29 | Cracking and degradation of bolts and studs from stress corrosion, fatigue, and boric acid corrosion. | NONE | NONE | Does the inspection frequency or fraction inspected need to be increased for extended life. | Further analysis needed. | Bolting degradation or failure in nuclear power plants<br>• Priority - HIGH<br>• Proposed inspecting 10% of bolts each outage, extending an outage 1.5 days. |
| (67) | | GSI 79 | Axial temperature gradients that could cause thermal stress in the flange area or in the studs. Cycling over the life of the plant may reduce fatigue margin or usage factor | NONE | NONE | Further analysis needed. | Further analysis needed. | Unanalyzed reactor vessel thermal stress during natural convection cooldown<br>• Priority -- MEDIUM |
| (68) | | GSI 94 | Major overpressure of the RCS, if combined with critical size crack, could result in a brittle failure of the reactor vessel. | NONE | NONE | Further analysis needed. | Further analysis needed. | Reactor Pressure Vessel Transient<br>• Priority -- RESOLVED with publication of NUREG-0224 and SRP 5.2<br>• Numerous reported pressure transients in PWRs where TS pressure and temperature limits were exceeded. Majority at solid water condition of startup or shutdown and relatively low reactor temperature. Less toughness at low temperature, therefore more susceptible to brittle fracture. |
| (70) | | USI A-49 | Neutron irradiation of the reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. | NONE | NONE | Further analysis needed. | Further analysis needed. | Pressurized Thermal Shock |
| (71) | | GSI B-6 | Through wall cracks and fatigue crack growth | NONE | NONE | Further analysis needed. | Further analysis needed. | Loads, load combinations, stress limits<br>• Priority -- HIGH<br>• See NUREG-2800 Supl |
| (72) | | GSI C-12 | Structural damage by vibrations of sufficient magnitude. Fretting due to deficient design and material selection for anti-vibration bars. Fatigue failure, particularly at the nozzle where stresses are highest. | NONE | NONE | Further analysis needed. | Further analysis needed. | Primary system vibrations<br>• Priority -- RESOLVED |

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| RPV (73) | LOW FLUX, LONG TIME IRRADIATION OF VESSEL AND EXTERNAL SUPPORTS | | | | | Time-temperature effects not understood. | Unknown at this time. | There are ACRS letters identifying the issue. The NRC is beginning to research the issue. There is evidence of larger $RT_{NDT}$ shifts in BWR PRVs than previously expected. It is not yet known whether low flux conditions are significant contributions to the shifts. |
| (74) | (Not used) | | | | | | | |
| (75) | | GSI A-11 | As plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial margins of safety. | NONE | NONE | Further analysis needed. | Further analysis needed. | Reactor vessel materials toughness<br>• Priority -- RESOLVED with issuance of NUREG-0744 and Generic Letter 82-66<br>• See A-49 for a related discussion. |
| (76) | | GSI A-26 | Toughness at low temperature | NONE | NONE | ? | Further analysis needed. | PRV transients<br>• Priority -- RESOLVED with publication of NUREG-0224 and SRP 5.2<br>• Numerous reported pressure transients in PWRs where TS pressure and temperature limits were exceeded. Majority at solid water condition of startup or shutdown and relatively low reactor temperature. Less toughness at low temperatures, therefore more susceptible to brittle fracture. |
| (77) | | GSI A-49 | Neutron irradiation of the reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. | NONE | NONE | Further analysis needed. | Further analysis needed. | Pressurized Thermal Shock |
| RPV (78) | CRACK INITIATION, STUD FAILURE, FLAW PROPAGATION (CISF) | Tech Spec 4.0.5 | References ASME Section XI. | Repair. | NONE | ? | Human factors of NDE should be evaluated; missed flaws or incorrectly sized should be avoided. Inspection criteria for new, irradiation induced failure modes that may occur may be needed. | Current flaw detection systems are not 100% effective. Detection systems may need to be improved. |
| (79) | | Tech Spec 3/4.4.10 | References ASME Section XI. | Data retention. Replacement. | NONE | Fluence effects not fully documented. | Documentation for data analysis needed in lieu of a replacement policy. | Bolt torque data may be collected and stress cycle analysis performed to justify life extension. |
| (80) | | SPP 4.5.2. III. 4 | Environmental control | NONE | NONE | N/A; limited to fabrication | Should the SRP address life extension? | Environmental conditions must be controlled during fabrication of austenitic stainless steels to reduce the possibility of sensitization and microfissuring. This section does not apply after the start of plant operations. |

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| (81) | CISF (contd) | SRP 5.2.1.2.II.2 | Inservice inspection code case and RG applicability. | NONE | NONE | Need to verify that the reference code cases cover an adequate time span for reactor operation. | NONE | NRC accepted code cases for ISI are found in RG 1.147 |
| (82) | | SRP 5.3.1.I.3 | Nondestructive examinations | NONE | NONE | ? | Should the SRP address life extension? | Special methods for NDE other than those in ASME should be reviewed. |
| (83) | | RG 1.150 | UT of RPV welds. | Monitoring | ? | Better flaw detection systems are needed for distributed micro-flaws | NDE human factors methods need evaluation for missed flaws or incorrectly sized flaws. Inspection criteria for new failure modes may be needed. | NDE of welds for crack initiation and growth. |
| (84) | | RG 1.65 | NDE/UT for crack, etc. in studs. | ISI | NONE | Revise RG 1.65 | Revise RG 1.65 | ISI of studs for damage with time. Updating of RG 1.65 is recommended. |
| (85) | | ASME Sec III NB-5000 | Acceptable flaws and benchmarking indications covered. | Adequately covers acceptable flaws and benchmarking indications. | Section under review for life extension applicability. | N/A | N/A | Flaws detected and benchmarked in Section III, and monitored and repaired by Section XI rules. This section does not apply after the start of plant operations. |
| (86) | | ASME Sec XI IWA-3300 IWA-3400 | Acceptable flaws and benchmarking indications covered. | Inspection and monitoring. | Section under review for life extension applicability. | N/A | N/A | Research or code work needed on the safety and risk improvement of inspection methods and time periods when the original design period is exceeded. Acceptable methods of establishing such guidelines and practices are needed. |
| (87) | | GSI 29 | Cracking and degradation of bolts and studs from stress corrosion, fatigue, boric acid corrosion and erosion corrosion. | NONE | NONE | Does the inspection frequency or fraction inspected need to be increased for extended life? | Further analysis needed. | Bolting degradation or failure in nuclear power plants<br>• Priority - HIGH<br>• Proposed inspecting 10% of bolts each outage, extending an outage 1.5 days. |
| (88) | | GSI 79 | Axial temperature gradients that could cause thermal stress in the flange area or in the studs. Cycling over the life of the plant may reduce fatigue margin or usage factor | NONE | NONE | Further analysis needed. | Further analysis needed. | Unanalyzed reactor vessel thermal stress during natural convection cooldown<br>• Priority -- MEDIUM |
| (89) | | GSI 94 | Major overpressure of the RCS, if combined with critical size crack, could result in a brittle failure of the reactor vessel. | NONE | NONE | Further analysis needed. | Further analysis needed. | Additional low temperature overpressure protection for LWRs<br>• Priority -- HIGH |
| (90) | | GSI 111 | Stress corrosion cracking | NONE | NONE | Further analysis needed. | Further analysis needed. | Stress corrosion cracking of pressure boundary ferritic steels in selected environments<br>• Priority -- LOW<br>• Classified as Licensing Issue |

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| (91) | CISF (contd) | GSI B-6 | Through wall cracks and fatigue crack growth | NONE | NONE | Further analysis needed. | Further analysis needed. | Loads, load combinations, stress limits<br>• Priority -- HIGH<br>• See NUREG-2800 Supl*** |
| (92) | | GSI C-12 | Structural damage by vibrations of sufficient magnitude. Fretting due to deficient design and material selection for anti-vibration bars. Fatigue failure, particularly at the nozzle where stresses are highest. | NONE | NONE | Further analysis needed. | Further analysis needed. | Primary system vibrations<br>• Priority -- RESOLVED |
| RPV (93) | CORROSION | | | | | | | This may be minor issue in RPVs. There is no evidence of internal problems. Leaks from external sources have caused corroding on the external surface. |
| (94) | | Tech Spec 3/4.4.7 | Chemistry control | N/A | NONE | NONE | N/A | Adequate programs exist for RCS chemistry control. |
| (95) | | SRP 4.5.2.I.2 | NONE | NONE | NONE | N/A | New life extension document needed beyond 40 years | Process controls are needed during manufacture and construction in order to lessen the likelihood of SCC. This section does not apply after the start of plant operations. |
| (96) | | SRP 4.5.2.II.4 | NONE | NONE | NONE | Update/modify the appropriate regulatory instruments. | Should the SRP address life extension? | Acceptance criteria for austenitic stainless steels referenced to RG 1.44 and 1.31. |
| (97) | | SRP 5.2.1.2.II.2 | Inservice inspection code case and RG applicability | NONE | NONE | Determine the adequacy of the three year time interval for code cases as related to ISI. | NONE | NRC accepted code cases are found in RG 1.147. |
| (98) | | SRP 5.2.4.II.4 | In service inspection intervals | NONE | NONE | Does inspection interval period need to be decreased in the last ten year interval? | New life extension document that reflects time greater than forty years? | Required inspection on the RPV pressure boundary during each 10 year interval of service is per Sec XI IWA-2000. Question is...should inspection Program B be used? |
| (99) | | SRP 5.3.1.I.1 | Material corrosion resistance. | NONE | NONE | Need assessment of changes in the properties and the corrosive environment as a function of time/fluence. | New life extension guidance needed beyond 40 years. | Adequacy for use of materials is to be assessed on the basis of mechanical and physical properties, effects of irradiation, corrosion resistance, and fabricability. This section does not apply after the start of plant operations. |
| (100) | | SRP 5.3.1.1.2 | NONE | NONE | NONE | N/A | New life extension guidance needed beyond 40 years. | The effects of special fabrication processes on stress corrosion characteristics must be evaluated. This section does not apply after the start of plant operations. |
| (101) | | SRP 5.3.1.1.3 | Nondestructive examinations | NONE | NONE | ? | New life extension document needed? | Special methods for NDE other than those in ASME should be reviewed. |

** Reactor Pressure Vessel

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (102) | CORROSION (contd) | SRP 5.3.1.I.7 | NONE | NONE | NONE | N/A | New life extension guidance needed beyond 40 years. | A design valuation of the materials properties of the reactor vessel fasteners is performed to ensure resistance to SCC. This section does not apply after the start of plant operations. |
| (103) | | SRP 5.3.1.III.2 | NONE | NONE | NONE | N/A | New life extension guidance needed beyond 40 years. | Special manufacturing processes are reviewed for effects on stress corrosion cracking. This section does not apply after the start of plant operations. |
| (104) | | SRP 5.3.1.III.4 | NONE | NONE | NONE | N/A | New life extension guidance needed beyond 40 years. | Special processes used on ferritic and austenitic stainless steels must be verified to conform to appropriate controls to avoid contamination and sensitization that can increase the possibility of SCC. This section does not apply after the start of plant operations. |
| (105) | | ASME Sec III NB-2160 NB-3121 | Corrosion allowances are provided. | NONE | Section under review for life extension applicability. | N/A | N/A | Intended to cover design, construction and acceptance testing of the pressure retaining components. The scope should be retained. It does not address aging or life extension. This section does not apply after the start of plant operations. |
| (106) | | ASME Sec XI IWA-2200 IWA-2300 IWA-2400 | Identifies the rules and requirements for inspection responsibilities, accessibility, examination methods, personnel qualifications frequency and records. Minimum basis for all requirements. | NONE | Section XI under review for applicability | N/A | N/A | Applies to the pressure retaining components only. The only life extension application is the design, fabrication, installation and inspection of replacement components. The scope of inspections may need to be broadened to cover areas that were not previously considered. All high stress areas need to be considered. Should be expanded to cover visual examination of the RPV exterior for corrosion. |
| (107) | | GSI 29 | Cracking and degradation of bolts and studs from stress corrosion, fatigue, boric acid corrosion and erosion corrosion. | NONE | NONE | Does the inspection frequency or fraction inspected need to be increased for extended life. | Further analysis needed. | Bolting degradation or failure in nuclear power plants<br>• Priority - HIGH<br>• Proposed inspecting 10% of bolts each outage, extending an outage 1.5 days. |
| (108) | | GSI 111 | Stress corrosion cracking | NONE | NONE | Further analysis needed. | Further analysis needed. | Stress corrosion cracking of pressure boundary ferritic steels in selected environments<br>• Priority -- LOW<br>• Classified as Licensing Issue |
| (109) | | GSI C-12 | Structural damage by vibrations of sufficient magnitude. Fretting due to deficient design and material selection for anti-vibration bars. Fatigue failure, particularly at the nozzle where stresses are highest. | NONE | NONE | NONE | Further analysis needed. | Primary system vibrations<br>• Priority -- RESOLVED |

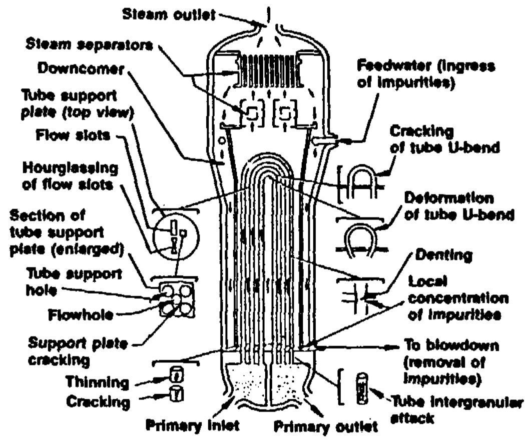
-- Reactor Pressure Vessel

# APPENDIX III

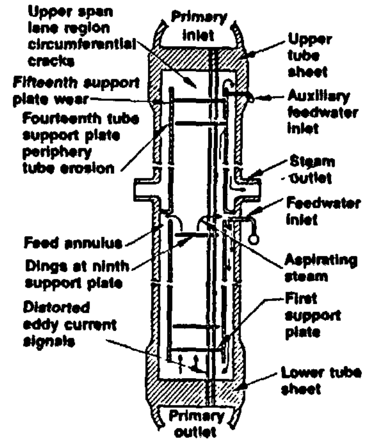
## REGULATORY INSTRUMENT REVIEW FOR STEAM GENERATOR

# Understanding and managing aging of PWR steam generator tubes

| | | |
|---|---|---|
| Materials | Tubes | - Inconel 600 or 690 |
| | Tube Sheet | - SA 508 clad with Ni-Cr-Fe alloy (equivalent to SB 168) |
| | Tube Supports | - SA 285 Gr. C Ferritic SS Type 405 or 409 |
| | Sleeves | - Inconel 625 or nickel bonded on outside surface of Inconel 600 or 690 |
| | Plugs | - Inconel 600 |
| Steam Generator Types | Recirculating | - Westinghouse, Combustion Engineering |
| | Once-Through | - Babcock & Wilcox |
| Stressors and Environment | Residual stresses, primary coolant chemistry (primarily hydrogen concentration), secondary coolant chemistry (chlorides, oxygen, copper, sulfates), phosphate chemistry, resin leakage from condensate polisher, brackish water, temperature, flow-induced vibrations, flow-velocities, and operating transients | |



Recirculating steam generator with indicated problem areas



Once-Through steam generator with indicated problem areas

## Understanding Aging (Materials, Stressors, and Environmental Interactions)

| Types | Sites | Aging Concerns |
|---|---|---|
| Recirculating Inside Surface | U-bends, roll transition, and dented regions | PWSCC (Pure water SCC) Tubes with low mill-annealing temperature are more susceptible |
| | Tube plugs | |
| Recirculating Outside Surface | Hot-leg tubes in tube-to-tubesheet crevice region | IGSCC, IGA |
| | Cold leg side in sludge pile or where scale containing copper deposits is found | Pitting |
| | Tubes in tube support regions | Denting |
| | Inadequately supported tube if dented near the top support plate | High-cycle fatigue |
| | Contact points between tube and antivibration bar | Fretting |
| | Tubes above tubesheet | Wastage |
| Once-Through Outside Surface | Tubes | Erosion-corrosion<br>Fatigue |
| | Tubes in upper tubesheet region | Environmental fatigue |

## Managing Aging

| Inservice Inspection, Surveillance, and Monitoring: NRC Requirements | Inservice Inspection, Surveillance, and Monitoring: Recommendations | Mitigation |
|---|---|---|
| Volumetric examination of hot leg side, U-bend portion, and (optionally) cold leg side of tubes in recirculating steam generators (IWB-2500)<br><br>Volumetric examination of the entire length of tubing in once-through steam generators (IWB-2500)<br><br>Frequency of inspection and number of tubes to be inspected (minimum of 3% of all tubes) are determined by Tech. Specs. (Reg. Guide 1.83)<br><br>Standards for allowable flaws in recirculating steam generators (standards for once-through steam generators are being prepared) (IWB-3521)<br><br>Flaw acceptance criteria determined by Tech Specs. (IWB-3630)<br><br>Criteria for determining necessity of plugging degraded tubes (Reg. Guide 1.121)<br><br>Unscheduled inservice inspection of each steam generator is required when primary to secondary tube leaks exceed the limits defined in Tech. Specs. | Follow Steam Generator Owners Groups' guidelines for continuous monitoring and control of secondary water chemistry<br><br>Reduce uncertainties in inspection results and quantify flaw growth rates<br><br>Monitor field performance of various sleeve designs<br><br>Perform inservice inspection of tube plugs | Prevent transient conditions in secondary water chemistry, install filters between condensate polishers and steam generators. Use ultrafiltration of makeup water and remedy condenser leakage as quickly as possible<br><br>Use shotpeening and rotopeening to introduce compressive residual stresses on tube inner surface in the roll transition region, and anneal U-bends to reduce PWSCC<br><br>Apply nickel plating on the inner surface of the tubes to prevent PWSCC crack initiation and propagation<br><br>Use tube rolling to eliminate tube sheet crevices and use crevice flushing, crevice alkalinity neutralization, alkaline impurity control, acid chloride elimination, hot soaks, sludge lancing, pressure pulse, water slap, chemical cleaning, and boric acid additions to control IGA/IGSCC<br><br>Eliminate copper pickup by use of titanium or stainless steel condenser tubes, and replace the copper-bearing alloys in the feedwater train to reduce pitting and denting<br><br>Use all-volatile treatment water chemistry, sludge lancing, chemical cleaning, hot soaks, hot blowdown and flushing, and elimination of hideout chemical concentration to control wastage<br><br>Use chemistry control to prevent concentration of impurities leading to fatigue crack initiation in once-through steam generators<br><br>Use lane-flow blocker in once-through steam generators to mitigate environmental fatigue |

## REGULATORY INSTRUMENT REVIEW FOR STEAM GENERATORS

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| STEAM GENERATOR (1) | STRESS: VESSEL PROBLEMS<br>* INTERGRANULAR STRESS CORROSION CRACKING--IGSCC<br>* WELDMENTS<br>* FATIGUE<br>* BOLTING<br>TUBE PROBLEMS<br>* IGSCC/SCC<br>* DENTING<br>* FATIGUE<br>* FRETTING<br>* CREVICES TUBE SHEET | CFRs<br>10 CFR 50, Appendix A | Records. | Records indicate maintenance problems. | None. | Material Evaluation-amount of coverage may need review to assure adequacy of inspection. | Time intervals and records adequacy. | References ASME Code ISI Requirements. |
| (2) | | Appendix B | Quality Assurance. | Proper QA will assure that adequate and accurate maintenance records are recorded and proper procedures are followed. | None. | Embrittlement Fatigue Stress. | Improved methods of detection for sizing cracks and IGA need to be reviewed/researched. | References ASME code for ISI requirements. |
| (3) | | TSs<br>3/4.4.5 | TSs requires inspection for wall penetrations (via any mode of wall thinning). | None, not addressed. | None. | Improved inspections for fretting/denting/IGSCC are needed. | Depends on the adequacy of Section XI, ISI. | Current inspection technique is eddy current testing of a fraction of tube bundle every 12-24 calendar months. No requirement concerning the SG shell or secondary side of the tube bundle. Root-cause analysis requirements for tube wall thinning are not explicitly delineated. Reporting requirements would have to be expanded. |
| (4) | | 3/4.4.10 | None. | None, not addressed. | None. | N/A | N/A--Section, TSs, 3/4.4.0.5. | This TS requires ASME, Section XI ISI, Inspections IWA - TSs 3/4.4.0.5. |
| (5) | | 3/4.4.0.5 (ASME, Section XI ISI, inspections are required.) | None. | None, not addressed. | None. | Depends on the adequacy of Section XI, ISI requirements/reports vs. aging needs. | Depends on the adequacy of Section XI, ISI requirements/reports vs. license renewal needs. | To expand the scope of the required inspections, the Section XI requirements would have to be changed or additional requirements would have to be added to TS 4.4.10 to specify the types/frequency of inspections and tests required. (The TSs currently cover the SG tube test frequency; Section XI doesn't address the frequency.) See Section XI analysis. |
| (6) | | 3/4.4.9.1.1 | Thermal/pressure cycles are recorded during RCS heat-up or cool-down (primary side only). | None, not addressed. | None. | Cycles are recorded. Might be useful in evaluating thermal/pressure cycle history/severity. | ? | Does not address rapid SG temperature transients caused by feedwater/emergency, feedwater transients, or rapid depressurizations of the secondary side. However, instances of transients or rapid depressurization would be reportable incidents. |

General Notes:
1. A "?" indicates further study/investigation is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessarily the aging issue.
3. For meaning of abbreviations, acronyms, and initialisms, used throughout, see acronyms on page xi, xii, and xiii of the report.

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (7) | STRESS (contd) | TSs (All) | | | Section XI SWG on operating plant criteria is developing several new (nonmandatory) appendices to address return to service situations when component TS have been exceeded. | | | |
| (8) | | SRPs 3.9.1 | General Design Criteria (GDC): design with sufficient margin. Also, GDC 1,2,14; 10 CFR 100, Appendix A and 10 CFR 50, Appendix B. | None (except for initial margins). | ? | None. | ? | Evaluate transients used in the design and fatigue analysis. |
| (9) | | 3.9.3 | Material Integrity. | References ASME Code. | None. | | | |
| (10) | | 5.2.3 | Review suitability of materials chosen. Perform tests (fracture toughness). Review procedures of manufacturing and welding. NDE per Section III, NB-2000. | None. | | None. | | Covers all reactor coolant pressure boundary. |
| (11) | | 5.2.4 | General. | None. | None. | Yes. | | ASME Sec XI requirements may not be adequate for deterioration. A new Appendix IV updates ET and may assist in these requirements. |
| (12) | | 5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME requirements for pressure boundaries). | None. | None. | None. | ? | Doesn't address specific aging issues; Section III, Class I component design criteria must be followed for pressure boundary. |
| (13) | | 5.4.2.1 | Overall Integrity References ASME. | None. | None. | | | |
| (14) | | 5.4.2.2 | References Regulatory Guide 1.83. | None. | In Rewrite. | Specific information needs to be complied with reference to a SG. | | New Guides needed to establish a deterioration overview of component. |
| (15) | | RGs 1.83 | | | NUPLEX Codes + Standards Work. Identify obvious changes--fatigue NUREG 0313. | Eliminate unintentional embedded 40-years constraint in NRC Regs and Codes and Standards. | | NUREG-0313 applies to containment boundary piping, not to SG tubes. |

-- Steam Generator

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|---|---|---|---|---|---|---|---|---|
| | STRESS (contd) | RGs (contd) | | | | | | |
| (16) | | 1.85 | No explicit reference to aging; provides an NRC acceptable list of approved ASME code cases for Section III. | The RG assists life extension by providing acceptance method of approved (ASME) materials acceptable to current approved Code Case. | RG revisions are made to include or exclude appropriate Code Case. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (17) | | 1.121 | Defines stress limits for flawed tubes. | None. | RG 1.121, Rev. 1, being developed from Steam Generator Integrity Program. | None. | None. | ASME code margins should be maintained. |
| (18) | | 1.147 | No explicit reference to aging; provides an NRC acceptable list of approved ASME code cases for Section XI. | The RG assists life extension by providing acceptance method of approved (ASME) inspection methods using current Code Cases. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (19) | | ASME Section III | | Requalification of components. | A new appendix is under consideration which may be used to regulate components which have exceeded the rules for cyclic operation in Section III, NB-3222.4. A Task Group has been formed to address the issue. This also applies to Section XI. | | Requalification rules for components. | From Attachment III of the Meeting Agenda for ASME-BNCS (Board Nuclear Codes and Standards) Steering Committee on PLEX (p. 8): Codes and standards such as BPV Section VIII for pressure vessels, B31.1 Code for Power Piping, and B16 Valve Standards fall within the jurisdiction of the ASME Board on Pressure Technology Codes and Standards (BPTCS). Both BNCS and BPTCS report to the ASME Council on Codes and Standards. Consideration of Section VIII and B31.1 will be under PLEX since they were extensively used in the construction of nuclear power plants prior to the publication of ASME B&PV Code, Section III. |
| (20) | | | Fatigue. | Fatigue curve revisions. | Section XI SWG on Operating Plant Criteria is reviewing fatigue curves in Section III to determine if they can be revised to accommodate operation beyond 40 years. | | Improved End-of-Life projections, 40 years and beyond. | Studies are under way to assess the magnitude of the effects of actual environmental conditions. Code curves are based on smooth specimens in air at room temperature, whereas materials in service have much rougher surfaces and are exposed to flowing coolant at operating temperatures. The actual surfaces are more prone to crack initiation. Studies and research to assess magnitude of effect of environmental factors could result in need for new Reg. Guide, Reg. Instruments and/or Code revisions. |

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|---|---|---|---|---|---|---|---|---|
| | STRESS (contd) | ASME (contd) | | | | | | |
| (21) | | Subsection NB NC NF | ASME Code provides requirements for design, e.g., pressure, temperature and mechanical loads, to assure the strength and integrity of the pressure boundary. | None explicit; yet provides guidance and sets forth rules for repairs and replacements of code components. | | Implied in the design design criteria (see comments). | Guidance is needed to cover the limits of component deterioration. | For Class I components, ASME Code requires special consideration for material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects and shall have provisions made for the effects during the design or specified life by a suitable increase in or addition to the thickness of the base metal over that determined by the design.<br><br>Rules cover design and construction requirements but do not cover deterioration. |
| (22) | | Section XI | Condition Assessment. | Requalification of systems and components. | Consideration of development of "new" Article IWX-8000, Requalification for "Fatigue Life." | Condition Assessment Information/Data. | Requalification of systems and components following expiration of operating license. | |
| (23) | | | TSs have been exceeded. | | SWG on Operating Plant Criteria developing several new nonmandatory appendices to address return to service situations when component TS have been exceeded. | | | |
| (24) | | IWB-3510 | Acceptance Standards. | Acceptance Standards. | None. | None. | Code evaluation limits for allowable indications of deterioration should be determined. | Present rules limit power but do not specify the number of defective tubes that a steam generator can contain before replacement is required. |
| (25) | | IWB-3512 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (26) | | IWB-3516 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (27) | | IWB-3519 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (28) | | IWB-3520 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (29) | | IWB-2500 | None. | None. | Regulatory Guide 1.83. | Yes. | Issue new Regulatory Guide 1.83. | 3% inspection is not adequate and full length is needed (tube sheet to tube sheet). Current rules require sequential ISI, if flaw(s) are detected, i.e. if something is found sample size is increased. |
| (30) | | IWB-3521 | Number of plugged or replaced tubes. | Number of plugged or replaced tubes. | | | Allowable indication for U-band flaw is now 40% through-wall. Growth rate suggests this is too high. | Number of tubes plugged in a steam generator could influence license renewal. |

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|---|---|---|---|---|---|---|---|---|
| | STRESS (contd) | ASME (contd) | | | | | | |
| (31) | | IWB-2413, IWB-2430, | None. | None. | | Yes. | | Should have increased examination program after detection of flaws: 3-1/2 years between examinations is now permitted. |
| (32) | | IWB-3630 | None. | | Rewrite of Regulatory Guide 1.83. | | Detection and sizing of SCC and IGA. | States that evaluation of cracks or IGA, etc., shall be performed by analysis acceptable to regulatory authority. |
| (33) | | Appendix IV | None. | None. | | | | Appendix IV rewrite in process.<br>This section of ASME (Section XI) may be transferred to Section V. Section XI will be replaced with a performance demonstration and qualification guideline. This is in the process of being written now but could take two years. |
| | | GSIs | | | | | | |
| (34) | | 18 | Failures of partially degraded steam generator tubes. | None. | None. | ? | None. | Steam-line Break with Consequential Small LOCA<br>* Implemented with TMI action plan Item 1.C.1 of NUREG-0737.<br>* Supplemented by NRC Generic Letter No. 82-33. |
| (35) | | 29 | Cracking and degradation of bolts and studs from stress corrosion, fatigue, boric acid corrosion, and erosion/corrosion. | None. | None. | Does the inspection frequency or fraction inspected need to be increased for extended life? | ? | Bolting Degradation or Failure in Nuclear Power Plants<br>* Priority - HIGH<br>* Proposed Inspecting 10% of bolts each outage, extending an outage 1.5 days. |
| (36) | | 35 | Loose objects in secondary piping could become missiles during steam generator blowdown and rupture one or more tubes. | None. | None. | Determine whether priority of the issue changes for longer plant life. | ? | Degradation of Internal Appurtenances in LWRs<br>* Related to B-60<br>* Priority - LOW. |
| (37) | | 66 | Mitigate or reduce steam generator tube degradations and ruptures. | None. | None. | ? | ? | Steam Generator Requirements<br>* Priority - HIGH. |
| (38) | | 67 | ? | None. | None. | ? | ? | Steam Generator Staff Actions<br>* Priority - LOW. |
| (39) | | 111 | Stress corrosion cracking in steam generator. | None. | None. | ? | ? | Stress corrosion cracking of pressure boundary ferritic steels in selected environments. |

-- Steam Generator

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|---|---|---|---|---|---|---|---|---|
| | STRESS (contd) | GSIs (contd) | | | | | | |
| (40) | | A-3 | Steam generator tube integrity degraded due to wastage and stress corrosion cracking. | None. | None. | ? | ? | Westinghouse Steam Generator Tube Integrity<br>* See NUREG-0371.<br>* Degradation decreased by conversion from phosphate to an all-volatile water treatment. Denting which leads to primary side stress corrosion continues to be a problem. |
| (41) | | A-4 | Steam generator tube integrity degraded due to wastage and stress corrosion cracking. | None. | None. | ? | ? | CE Steam Generator Tube Integrity<br>* See NUREG-0371.<br>* Degradation decreased by conversion from phosphate to an all-volatile water treatment. Denting which leads to primary-side stress corrosion continues to be a problem. |
| (42) | | A-5 | ? | None. | None. | ? | ? | B&W Steam Generator Tube Integrity<br>* See NUREG-0371. |
| (43) | | A-12 | Lamellar tearing and low fracture toughness of steam generator and reactor pump support materials. | None. | None. | ? | ? | Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports<br>* Solution made available in October 1983 with publication of NUREG-0577.<br>* Applies to new construction when SRP, Section 5.3.4 is issued. |
| (44) | | A-15 | Operation of LWR results in slow corrosion of interior metal surfaces of the primary coolant system. These are activated by neutron flux when circulated through reactor and plate-out. | None. | None. | ? | ? | Primary coolant system decontamination and steam generator chemical cleaning. |
| (45) | | USIA-47 | Define generic criteria for plant specific reviews for steam generator overfill transients in PWRs and reactor overfills in BWRs. | | | | | Safety Implications of Control Systems<br>* Need to address rupture of instrument sensing lines. |
| (46) | | B-60 | Presence of loose object in primary coolant system can be indicative of degraded reactor safety system resulting from failure or weakening component. | None. | None. | ? | Perform trend analysis on loose part monitoring data. | Loose Part Monitoring System<br>* Priority - RESOLVED. |

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|---|---|---|---|---|---|---|---|---|
| | STRESS (contd) | ANS | | | | | | |
| (47) | | 3.2 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | Quality Assurance documentation and/or records are required to show and evaluate deterioration of a component. |
| (48) | | 5.37 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |
| (49) | | 5.3.10 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |
| (50) | DYNAMIC EFFECTS<br>* VIBRATION<br>* THERMAL CYCLES<br>* EROSION | CFRs<br>10 CFR 50, Appendix A | Records. | Yes. | None. | Material Evaluation. | Interval between exams need to be reviewed. | References ASME Code for ISI. |
| (51) | | Appendix B | Quality Assurance. | Yes. | None. | Embrittlement Fatigue Stress. | Methods of detecting and sizing cracks and IGA need to be reviewed. | References ASME Code for ISI. |
| | | TSs | | | | | | |
| (52) | | 3/4.4.5 | TS requires eddy current testing of tubes that could indicate tube erosion (shell not addressed). | None, not addressed. | None. | Inspect secondary side of tubes for vibration (fretting) and shell erosion). | Reports from inspections could be used to evaluate tube erosion conditions, though root-cause analysis may be lacking. | Eddy current testing finds wall thinning but not necessarily the mode of thinning unless the utility included this in its corresponding report of the inspection to the NRC. Only a fraction of the total number of tubes are required to be inspected during an outage. |
| (53) | | 3/4.4.10<br>4.0.5<br>ASME Code Class Maintenance Committee | None. | None, not addressed. | None. | N/A | Depends on adequacy of Section XI ISI requirements and other requirements already in place. | TS 4.4.10 defers to T.S. 4.0.5 which specifies that the ASME, Section XI, requirements are applicable. See Section XI analysis. |
| (54) | | 3/4.4.9.1.1<br>RCS heat-up/ cool-down | Thermal cycles are recorded during RCS heat-up and cool-down. | None, not addressed. | None. | The thermal cycles are recorded and retained. Perhaps useful in evaluating cycle history/severity. | ? | Does not address rapid SG temperature transients caused by feedwater/emergency feedwater transients (e.g., transients associated with feedwater/emergency feedwater SG over-feeding or rapid loss at feedwater preheating). |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (55) | DYNAMIC EFFECTS (contd) | SRPs<br>3.9.2 | Review of the criteria, testing procedures, and dynamic analysis. | None. | ? | None. | ? | Assure structural and functional integrity under vibratory loading. |
| (56) | | 3.9.3 | Material integrity. | Ref. ASME. | None. | Complete and reliable data base. | Complete and reliable data base. | |
| (57) | | 5.2.4 | General. | None. | None. | Yes. | | ASME Sec XI requirements may not be adequate for deterioration. |
| (58) | | 5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME Section III requirements for pressure boundaries). | None. | None. | None. | ? | Doesn't address specific issues; Section III must be followed. |
| (59) | | 5.4.2.1 | Overall integrity, References ASME. | None. | None. | | | |
| (60) | | 5.4.2.2 | References Regulatory Guide 1.83. | None. | In Rewrite. | Yes. | | New Guides needed to establish a deterioration overview of component. |
| (61) | | RGs<br>1.121 | Defines dynamic loading considerations governing tube failure. | Limited. | RG 1.121 limited to tube integrity. | Consideration of other structural degradation, i.e., vessels. | Long-term failure resistance of SG vessel. | |
| (62) | | 1.147 | No explicit reference to aging; provides an NRC acceptable list of approved ASME code cases for Section XI. | The RG assists life extension by providing acceptance method for approved (ASME) inspection methods using current Code Cases. | RG revisions are made include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (63) | | 1.85 | No explicit reference to aging; provides an NRC acceptable list of approved ASME code cases for Section III. | The RG assists life extension by providing acceptance method for approved (ASME) materials acceptable to current approved Code Cases. | RG revisions are made include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (64) | | ASME<br>Section III, Subsection<br>NB<br>NC<br>NF | ASME Code provides requirements for design, e.g., pressure, temperature and mechanical loads, to assure the strength and integrity of the pressure boundary. | None explicit; yet, provides guidance and sets forth rules for repairs and replacements of code components. | | Implied in the design design criteria (see comments). | Guidance is needed to cover the limits of component deterioration. | For Class I components, ASME Code requires special consideration for material subject to thinning by corrosion, erosion, mechanical abrasion or other environmental effects and shall have provisions made for the effects during the design or specified life by a suitable increase in or addition to the thickness of the base metal over that determined by the design.<br><br>Rules cover design and construction requirements but do not cover deterioration. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (65) | DYNAMIC EFFECTS (contd) | ASME (contd) Section XI | Corrosion and Erosion. | | A new SWG on pipe wall-thinning has been established and will review this issue with EPRI. Main area of concern at this time is the effects of erosion/corrosion on piping in non-nuclear applications ANSI/ASME B31G-1984, <u>Manual for Determining the Remaining Strength of Corroded Pipelines</u>. | | | The Section XI Subgroup on Nondestructive Examination is currently developing a "Code Case" and nonmandatory Appendix on Ultrasonic Detection and Measurement of Erosion/Corrosion.<br><br>ANSI/ASME B-31.G may have a limited application to nuclear facilities. |
| (66) | | IWB-2500 | None. | None. | Rewrite of Regulatory Guide 1.83. | Yes. | Issue new Regulatory Guide 1.83. | 3% inspection is not adequate and full length is needed (tube sheet to tube sheet). Current rules require sequential ISI, if flaw(s) are detected, i.e., if something is found, sample size is increased. (The tube sampling is set by the TSs, not by Section XI.) |
| (67) | | IWB-3510 | Acceptance Standards. | Acceptance Standards. | None. | None. | Evaluation and guides for allowable indications and deterioration limits should be determined. | Present rules limit power but do not specify the number of defective tubes that a steam generator can contain before replacement is necessary. |
| (68) | | IWB-3512 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (69) | | IWB-3516 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (70) | | IWB-3519 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (71) | | IWB-3520 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (72) | | IWB-3521 | Number of plugged or replaced tubes. | Number of plugged or replaced tubes. | | | Allowable indication for U-bend flaw is 40% through-wall; growth rate suggests this is too high. This may not be conservative enough, i.e., all indications of a flaw should require plugs. | Number of tubes plugged in a steam generator could influence license renewal. |

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|---|---|---|---|---|---|---|---|---|
| | DYNAMIC EFFECTS (contd) | ASME (contd) | | | | | | |
| (73) | | IWB-2413, 2430 | None. | None. | | Yes. | | Should have increased examination program after detection of flaws: 3-1/2 years between examinations is now possible. |
| (74) | | IWB-3630 | None. | | Regulatory Guide 1.83 | | Yes, detection and sizing of SCC and IGA. | States that evaluation of cracks or IGA, etc., shall be performed by analysis acceptable to regulatory authority. |
| (75) | | Appendix IV | None. | None. | | | | Appendix IV rewrite in process.<br><br>This section of ASME (Section XI) may be transferred to Section V. Section XI will be replaced with a performance demonstration and qualification guideline. This is in the process of being written now but could take two years. |
| | | GSIs | | | | | | |
| (76) | | 18 | ? | None. | None. | ? | ? | Steam-line break with consequential small LOCA. |
| (77) | | 35 | Loose objects in secondary piping could become missiles during steam generator blowdown and rupture one or more tubes. Presence of loose objects implies a system has failed. | None. | None. | ? | ? | Degradation of Internal Appurtenances in LWRs<br>* Priority - LOW. |
| (78) | | 60 | Lamellar tearing represents a reduction in load a joint could handle, particularly under emergency conditions such as seismic conditions. | None. | None. | ? | ? | Lamellar Tearing of Reactor System Structural supports<br>* Is a subtask under A-12. |
| (79) | | 66 | Mitigate or reduce steam generator tube degradations and ruptures. | None. | None. | ? | ? | Steam Generator Requirements<br>* Four Items Priority - HIGH. |
| (80) | | 67 | ? | None. | None. | ? | ? | Steam Generator Staff Actions<br>* Priority - LOW. |
| (81) | | A-3 | None. | None. | None. | ? | ? | Westinghouse Steam Generator Tube Integrity<br>* See NUREG-0371. |

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|---|---|---|---|---|---|---|---|---|
| (82) | DYNAMIC EFFECTS (contd) | GISs (contd) A-4 | None. | None. | None. | ? | ? | CE Steam Generator Tube Integrity<br>* See NUREG-0371. |
| (83) | | A-5 | Tube cracks of unknown origin propagated in the circumferential direction by flow-induced vibrations. Erosion cavitation phenomena. | None. | None. | ? | ? | B&W Steam Generator Tube Integrity<br>* See NUREG-0371. |
| (84) | | A-12 | Lamellar tearing and low-fracture toughness of steam generator and reactor pump support materials. | None. | None. | ? | ? | Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports<br>* Solution made available in October 1983 with publication of NUREG-0577.<br>* In the case of North Anna, licensee agreed to raise temperature above 225°F before pressurizing above 1000 PSI. |
| (85) | | USIA-47 | Contribution of control system failures resulting in overfilling transients and pressurized thermal shock. | None. | None. | ? | ? | Safety Implications of Control Systems. |
| (86) | | B-60 | Presence of loose object in primary coolant system can be indicative of degraded reactor safety system resulting from failed or weakened component. | None. | None. | ? | ? | Loose Part Monitoring System<br>* Priority - RESOLVED. |
| (87) | | ANS 3.2 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | Quality Assurance documentation and/or records are required to show and evaluate deterioration of a component. |
| (88) | | 5.37 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |
| (89) | | 5.3.10 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |

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|---|---|---|---|---|---|---|---|---|
| (90) | STRUCTURAL INTEGRITY * FRACTURE TOUGHNESS * EMBRITTLEMENT THERMAL * SEISMIC (DAMAGE) FAILURES | CFRs 10 CFR 50, Appendix A | Records. | Yes. | None. | Material Evaluation. | Intervals between exams need to be reviewed. | References ASME for ISI. |
| (91) | | Appendix A, Criterion 2 | Design base for protection of equipment from natural phenomena including components important to safety. | None explicit; provides direction to use most severe known historical phenomena for the design, i.e., this conservative design base. | None. | Generic and specific design(s) must meet current and known natural phenomena. | Design must meet current design criteria for the phenomena. | Regulation should be updated to reflect life extension options and the option to evaluate nuclear facilities for plant aging. |
| (92) | | Criterion 51 | None explicit/ stipulates Reactor Pressure Boundary to be designed to include considerations (sufficient margins/factor of safety) for pressure containment under operating, testing, and postulated accident conditions such that the pressure boundary ferritic materials behave in a nonbrittle manner. | None explicit; implied by the design if the ferritic material continues to show non-brittle properties. | None. | Methods to measure embrittled properties. | Improved NDE techniques to determine embrittlement properties. | Regulation criteria needs updating to feature life extension applications. |
| (93) | | Appendix B | Quality Assurance. | Yes. | None. | Embrittlement Fatigue Stress. | Provide audits/records of methods of detection and sizing cracks and IGA. | References ASME Code for ISI requirements. |
| (94) | | TSs 3/4.4.5 | N/A | None, not addressed. | None. | N/A | ? | TS requires periodic eddy current testing of SG U-tubes. |
| (95) | | 3/4.4.10 | None. | None, not addressed. | None. | N/A | ? | This TS requires ISI ASME, Section IX. If Section XI is not adequate, and the TSs are to be expanded, this may be where the revisions should occur. |
| (96) | | 3/4.4.7.9 | Requires periodic testing of snubbers associated with the SG and piping. | None, not addressed. | None. | | | This TS is limited with reference to the seismic issue, i.e., seismic monitoring of the site is required. No other references are made. In addition, periodic testing of the SG snubbers does not occur. The utilities are unable to do so. |

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|---|---|---|---|---|---|---|---|---|
| (97) | STRUCTURAL INTEGRITY (contd) | TSs (all) | | | Working Group on Operating Plant Criteria, a proposed addition to Article IWB-3000 of Section XI. Would contain rules and formulas for performing an evaluation of a component's fitness for continued service when either TSs or elements of its construction/design limitations had been exceeded. | | | |
| (98) | | SRPs 3.9.3 and 3.9.3 Appendix A | Material Integrity Reference to ASME Section III and GDC 1,2,4,14,15. | Reference ASME. None (except for original design margins). | None. | None. | | Loading combinations, system operating transients, and stress limits. |
| (99) | | 5.2.3 | Review suitability of materials chosen. Perform tests (fracture toughness). Review procedures of manufacturing and welding. NDE testing as per Section III, NB-2500. | None. | None. | | | Covers all reactor coolant pressure boundary. |
| (100) | | 5.2.4 | Require ISI Program of RCPB (GDC-32) to assess structural and leak-tight integrity. | None. | ? | ? | ? | Based on requirements of 10 CFR 50.55a and detailed in ASME, Section XI. |
| (101) | | 5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME Section III requirements for pressure boundaries). | None. | None. | None. | ? | Doesn't address specific issues; Section III must be followed. |
| (102) | | 5.4.2.1 | Overall Integrity Meet GDC 1,14,15,31 and 10 CFR 50, Appendix A. Have sufficient design margin. Review selection and fabrication of materials. Must meet ASME code. | None. | None. | | | To meet the GDC 1, the acceptable materials are specified in Section III, Appendix I, Section II, Parts A,B,C. Must also meet RG 1.85. To meet GDC 1,14, and 31, the fracture toughness must meet 10 CFR 50, Appendix G, and Section III NB-2300 and ASME, Appendix G G-2000. For welding, see ASME Code Sections III and IX. |
| (103) | | 5.4.2.2 | Reference, Regulatory Guide 1.83. | None. | Rewrite in process. | Yes. | | New guides needed to establish a deterioration overview of component. |

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|---|---|---|---|---|---|---|---|---|
| (104) | STRUCTURAL INTEGRITY (contd) | RGs 1.121 | Defines methods for calculating minimum acceptable tube wall thickness. | Limited. | RG 1.121 limited to tube integrity. | Consideration of other structural degradation, i.e., vessels. | Long-term failure resistance of SG vessel. | |
| (105) | | 1.147 | No explicit reference to aging; provides an NRC acceptable list of approved ASME code cases for Section XI. | The RG assists life extension by providing acceptance method for approved (ASME) inspection methods using current Code Cases. | RG revisions are made include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (106) | | 1.85 | No explicit reference to aging; provides an NRC acceptable list of approved ASME code cases for Section III. | The RG assists life extension by providing acceptance method for approved (ASME) materials acceptable to current approved Code Cases. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (107) | | ASME Section III, Subsection NB NC NF | ASME Code provides requirements for design, e.g., pressure, temperature and mechanical loads, to assure the strength and integrity of the pressure boundary. | None explicit; yet provides guidance and and sets forth rules for repairs and replacements of code components. | | Implied in the design criteria (see comments). | Guidance is needed to cover the limits of component deteriortion. | For Class I components, ASME Code requires special consideration for material subject to thinning by corrosion, erosion, mechanical abrasion or other environmental effects and shall have provisions made for the effects during the design or specified life by a suitable increase in or addition to the thickness of the base metal over that determined by the design.<br><br>Rules cover design and construction requirements but do not cover deterioration. |
| (108) | | IWB-3510 | Acceptance Standards. | Acceptance Standards. | None. | None. | Evaluation and Guide for allowable indications and deterioration limits should be determined. | Present guides limit power but do not specify the number of defective tubes that a steam generator can contain before replacement is required. |
| (109) | | IWB-3512 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (110) | | IWB-3516 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |

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|---|---|---|---|---|---|---|---|---|
| (111) | STRUCTURAL INTEGRITY (contd) | ASME (contd) IWB-3519 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (112) | | IWB-3520 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (113) | | IWB-2500 | None. | None. | Regulatory Guide 1.83. | Yes. | Issue new Regulatory Guide 1.83. | |
| (114) | | IWB-3521 | Number of plugged or replaced tubes. | Number of plugged or replaced tubes. | | | Allowable indication for U-bend flaw is now 40% through-wall. Growth rate suggests this is too high. | Number of tubes plugged in a steam generator could influence license renewal. |
| (115) | | IWB-2413, 2430 | None. | None. | | Yes. | | Should have increased examination program after detection of flaws: 3-1/2 years between examinations is now possible. |
| (116) | | IWB-3630 | None. | | Regulatory Guide 1.83. | | Yes, detection and sizing of SCC and IGA. | States that evaluation of cracks or IGA, etc., shall be performed by analysis acceptable to regulatory authority. |
| (117) | | Article IV | None. | None. | | | | Regulatory Authority has been rewritten and is in final stage.<br>This section of ASME (Section XI) may be transferred to Section V. Section XI will be replaced with a performance demonstration and qualification guideline. This is in the process of being written now but could take two years. |
| (118) | | ANS 3.2 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | Quality Assurance documentation and/or records are required to show and evaluate deterioration of a component. |
| (119) | | 5.37 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |
| (120) | | 5.3.10 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |

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|---|---|---|---|---|---|---|---|---|
| (121) | CORROSION<br>* WATER CHEMISTRY<br>* IMPURITIES<br>* INTERGRANULAR ATTACK (IGA)<br>* IRRADIATION ENHANCED EFFECTS (INCREASED RADIOLYTIC DECOMPOSITION OF WATER AND CONDENSER INLEAKAGE INFLUENCES WATER CHEMISTRY) | CFRs<br>10 CFR 50 10, Appendix A | Records. | Yes. | None. | Material Evaluation. | Interval between exams needs to be reviewed. | References ASME for ISI requirements. |
| (122) | | Appendix B | Quality Assurance. | Yes. | None. | Embrittlement Fatigue Stress. | Methods of detection and sizing for cracks and IGA need to be revised. | |
| (123) | | TSs<br>3/4.4.5 | Eddy current testing could detect wall thinning due to chemistry problems. | None, not addressed. | None. | Inspection reports could be expanded to look at SG materials for chemistry-specific issues. | Inspection reports might be useful for renewal criteria review if requirements for reports were made more specific. | This TS requires eddy current testing of SG U-tubes. IGA effects on U-tube wall thickness could be detected though the cause is not necessarily inferred. Reporting requirements would have to be expanded. |
| (124) | | 3/4.4.10<br>4.05 | Requires ASME, Section XI, requirements for SGs. | None, not addressed. | None. | N/A | Depends on whether reports/records of ASME, Section XI, are adequate to be useful. | This TS requires the requirements of TS 3/4.4.0.5 (ASME, Section XI, ISI) be applied to SG. |
| (125) | | 3/4.4.4.7 | Requires primary coolant (RCS) chemistry control. | None, not addressed. | None. | N/A | Primary side chemistry analysis records are retained. Perhaps useful in evaluating SG history with regard to chemistry of fluid (RCS). | GENERAL CORROSION NOTE: Technical Specifications do not address secondary (steam plant side) chemistry control at all. This is where chemical attack of U-tubes would seem the most likely. Therefore, especially for old SGs, secondary chemistry and its bearing on U-tube integrity would be of great importance. (Secondary side chemistry is administratively controlled but not by TSs)<br>Limits placed on Cl, F, and dissolved oxygen. Are these adequate? |

| Parameter | Steady State Limit | Transient Limit |
|---|---|---|
| (dissolved $O_2$)* | <0.10 ppm | <1.00 ppm |
| ($Cl^-$) <0.15 ppm | <1.50 ppm | |
| ($F^-$) <0.15 ppm | <1.50 ppm | |

*N/A if RCS temp. is <250°F.

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|---|---|---|---|---|---|---|---|---|
| (126) | CORROSION (contd) | SRPs 3.9.3 | Material Integrity. | References ASME. | None. | | | |
| (127) | | 5.2.3 | Corrosion. | None. | None. | | Yes. | Covers construction only. |
| (128) | | 5.2.4 | General. | None. | None. | Yes. | | ASME Sec XI requirements may not be adequate for deterioration. |
| (129) | | 5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME Section III requirements for pressure boundaries). | None. | None. | None. | ? | Doesn't address specific issues; Section III must be followed. |
| (130) | | 5.2.4.1 | Must meet GDC 1,14,15, and 31 of 10-CFR-50, Appendix A. The design must have sufficient margin. Requires review selection and fabrication of materials. Fracture prevention criteria must be satisfied. | None. | | None. | ? | To meet the GDC 1, the acceptable materials are specified in Section III, Appendix I, Section II, Parts A,B,C. Must also meet Regulatory Guide 1.85. For welding, see Sections III and IX. To meet corrosion-resistance, see Section IX, Part QW, Articles I, II, III and IV, Reference BTP MTEB 5-3 (Water Chemistry for Steam Generators). |
| (131) | | BTP METB 5-3 | Maintain water chemistry. Guides to design to prevent sludge build-up. | Maintaining "good" water chemistry. | ? | ? | ? | Actually mentions "long-term reliable operation." Reference Regulatory Guide 1.37. |
| (132) | | 5.4.2.2 | Reference Guide 1.83. | None. | Rewrite in process. | Yes. | | New Guides need to establish a deterioration overview of component. |
| (133) | | 10.4.8 | Steam generator blowdown system review design basis. GDC 14; secondary water chemistry. | None. | ? | None. | ? | Review only initial design of the system. |
| (134) | | RG 1.121 | Degradation growth allowance included in minimum tube wall determination. | Limited. | Revision of RG 1.121 partially addresses degradation growth allowances for tubes. | Better definition of degradation growth rates needed for specific defects. | Better models of degradation growth needed to protect remaining tube integrity. | Available data from accelerated laboratory tests. May not be useful for predicting SG performance. |
| (135) | | ASME Section III, Subsection NB NC NF | ASME Code provides requirements for design, e.g., pressure, temperature and mechanical loads, to assure the strength and integrity of the pressure boundary. | None-explicit; yet provides guidance and sets forth rules for repairs and replacements of code components. | | Implied in the design design criteria (see comments). | Guidance is needed to cover the limits of component deterioration. | For Class I components, ASME Code requires special consideration for material subject to thinning by corrosion, erosion, mechanical abrasion or other environmental effects and shall have provisions made for the effects during the design or specified life by a suitable increase in or addition to the thickness of the base metal over that determined by the design. <br><br> Rules cover design and construction requirements but do not cover deterioration. |

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|---|---|---|---|---|---|---|---|---|
| | CORROSION (contd) | ASME (contd) | | | | | | |
| (136) | | Section XI IWB-3510 | Acceptance Standards. | Acceptance Standards. | None. | None. | Evaluation and Guide for allowable indications and deterioration limits should be determined. | Present guides limit power but do not specify the number of defective tubes that a steam generator can contain before replacement is required. |
| (137) | | IWB-3512 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (138) | | IWB-3516 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (139) | | IWB-3519 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (140) | | IWB-3520 | Acceptance Standards. | Acceptance Standards. | None. | None. | | |
| (141) | | IWB-2500 | None. | None. | Regulatory Guide 1.83. | Yes, Guide 1.83. | Issue new Regulatory Guide 1.83. | 3% inspection is not adequate and full length is needed (tube sheet to tube sheet). Current rules require sequential ISI, if flaw(s) are detected. i.e., if something is found, sample size is increased. |
| (142) | | IWB-3521 | Number of tubes plugged. | Number of tubes plugged. | | | Allowable indication for U-bend flaw is now 40% thru wall; growth rate suggests this is too high. | Amount of tubes plugged in a steam generator could influence license renewal. |
| (143) | | IWB-2413, 2430 | None. | None. | | Yes. | | Should have increased examination program after detection of flaws: 3-1/2 years between examinations is now possible. |
| (144) | | IWB-3630 | None. | | Regulatory Guide 1.83. | | Detection and sizing of SCC and IGA. | States that evaluation of cracks or IGA, etc., shall be performed by analysis acceptable to regulatory authorization. |
| (145) | | Appendix IV | None. | None. | | | | Appendix IV rewrite in process.<br>This section of ASME (Section XI) may be transferred to Section V. Section XI will contain performance demonstration and qualification rules. This could take two years. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | CORROSION (contd) | GSIs | | | | | | |
| (146) | | 18 | ? | None. | None. | ? | ? | Steam-line break with consequential small LOCA. |
| (147) | | 35 | ? | None. | None. | ? | ? | Degradation of Internal Appurtenances in LWRs<br>* Priority - LOW. |
| (148) | | 66 | Mitigate or reduce steam generator tube degradations and ruptures. | None. | None. | ? | ? | Steam Generator Requirements<br>* Four Items Priority - HIGH. |
| (149) | | 67 | ? | None. | None. | ? | ? | Steam Generator Staff Actions<br>* Priority - LOW. |
| (150) | | 111 | Stress corrosion cracking in steam generator. | None. | None. | ? | ? | Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments. |
| (151) | | A-3 | Steam generator tube integrity degraded due to wastage and stress corrosion cracking. | None. | None. | ? | ? | Westinghouse Steam Generator Tube Integrity<br>* See NUREG-0371.<br>* Degradation decreased by conversion from phosphate to an all-volatile water treatment. Denting which leads to primary side stress corrosion continues to be a problem. |
| (152) | | A-4 | Steam generator tube integrity degraded due to wastage and stress corrosion cracking. | None. | None. | ? | ? | CE Steam Generator Tube Integrity<br>* See NUREG-0371.<br>* Degradation decreased by conversion from phosphate to an all-volatile water treatment. Denting which leads to primary side stress corrosion continues to be a problem. |
| (153) | | A-5 | ? | None. | ? | ? | ? | B&W Steam Generator Tube Integrity<br>* See NUREG-0371. |
| (154) | | A-12 | Lamellar tearing and low fracture toughness of steam generator and reactor pump support materials. | None. | ? | ? | ? | Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports<br>* Solution made available in October 1983 with publication of NUREG-0577. |
| (155) | | A-15 | Operation of LWR results in slow corrosion of interior metal surfaces of the primary coolant system, then activated by neutron flux when circulating through the reactor and plate-out. Once plated, they cannot be removed by plant water chemistry system. | None. | None. | Assess the efficiency of water chemistry system to meet extended plant life. | ? | Primary coolant system decontamination and steam generator chemical cleaning. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (156) | CORROSION (contd) | GSIs (contd) B-60 | Presence of loose object in primary coolant system can be indicative of degraded reactor safety system resulting from failure or weakening component. | None. | None. | ? | ? | Loose Part Monitoring System<br>* Priority - RESOLVED. |
| (157) | | ANS 3.2 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | Quality Assurance documentation and/or records required to show and evaluate deterioration of a component. |
| (158) | | 5.37 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |
| (159) | | 5.3.10 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |
| (160) | SURVEILLANCE<br>* TESTING<br>* INSPECTION<br>* MAINTENANCE & REPAIR<br>* TUBE CLEANING & REPAIR<br>* NDE<br>* INFORMATION/DATA REQUIREMENTS<br>- FREQUENCY<br>- DATA COLLECTION<br>- DATA EVALUATION<br>- TREND CURVES<br>- RECORDING KEEPING | CFRs 10 CFR 50 | Surveillance Program | "New programmatic criteria" standard that could, by Ref., become a part of 10 CFR 50 if it were established by a standard's group at request of NUPLEX. | "Programmatic criteria" type standard being considered by NUPLEX for entire nuclear plant and all components of concern. | Overall guidance/ planning for surveillance program. | This standard would serve as a guide to a nuclear utility in become their devel-plant-specific program plan. It would provide overall guidance for steam generator issues. | Suggested that this "Programmatic-Criteria" standard become a parent standard and that other daughter standards may be needed to specify inspection techniques, the evaluation and identification of aging mechanisms, the establishment of special or enhanced maintenance programs, and the development and implementation of recordkeeping and trending programs. Guidance could also be provided seperately for evaluating the effects of plant transient events on component or system life. This standard is expected to be used by nuclear utilities to establish a plan for assessing the feasibility of a plant life extension program and for developing and implementing such a program where practical. It would assist utility personnel in identifying actions required to preserve a plant life extension option and for identifying needs for equipment testing, inspection, monitoring, and assessment. It will provide a framework for addressing future regulatory requirements as they might affect the extension of plant life and to support particular plant operating and maintenance practices as an integral part of prudent plant management.<br><br>(Prepared by J. Christie for NUPLEX Codes and Standards Subcommittee.) |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | CFRs (contd) | | | | | | |
| (161) | | | Could lead to NRC rule changes. | | NUPLEX Codes and Standards Identification Number PWR.08 on Steam Generator (see comments). NUPLEX IR to be prepared. | | | * Description: Develop UT techniques and ISI program for detection and sizing of flaws in steam generators. |
| | | TSs | | | | | | |
| (162) | | 3.4.4.5 | Requires eddy current testing of SG U-tubes. | None, not addressed. | None. | N/A | ? | TS requires eddy current testing of SG U-tubes. IGA effects on U-tube wall thickness could be detected though the cause is not necessarily inferred. Reporting requirements would have to be expanded. |
| (163) | | 3/4.4.10<br>3/4.0.5 | Requires ASME, Section XI, treatment for SG with regard to maintenance if Code Class 1, 2, or 3. | None, not addressed. | None. | N/A | ? | This TS requires the requirements of TS 3/4.4.0.5 (ASME Section XI, ISI) to be applied to the Steam Generator. |
| | | SRPs | | | | | | |
| (164) | | 3.9.3 | Material Integrity. | Reference ASME Codes. | None. | | | |
| (165) | | 5.2.3 | Corrosion and Stress. | None. | None. | | Yes. | Covers construction only. |
| (166) | | 5.2.4 | General. | None. | None. | Yes. | | ASME Sec XI requirements may not be adequate for deterioration. |
| (167) | | 5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME Section III requirements for pressure boundaries). | None. | None. | None. | ? | Doesn't address specific issues; Section III must be followed. |
| (168) | | 5.4.2.1 | Overall Integrity, Reference ASME Codes. | None. | None. | | | |
| (169) | | 5.4.2.2 | Steam Generator Tube In-service Inspection (GDC-32), Reference to RG 1.83 and TS for each NSSS supplier (NUREG-0103, 0212, 0452). | ISI | ? | ? | ? | Inspection Program to detect aging; is it effective? Doesn't prevent aging. |
| (170) | | 6.6 | In-service Inspection of Class 2 & 3 components (SG secondary side.) | None. | ? | ? | ? | Program based on 10 CFR 50.55a and detailed in Section XI. |
| (171) | | 10.3.6 | NDE and cleaning (Steam Generator shell side) with reference to RG 1.37 for cleaning. Also 10 CFR 50, 50.55a, and Appendix A. | None. | ? | None. | ? | Concerned with Class 2 & 3 components, therefore only shell side of SG. Mainly references other documents to follow. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (172) | SURVEILLANCE (contd) | SRPs (contd) 17.2 | Implement QA program. | None. | ? | None. | | SRP does not provide an in-depth description of QA requirements and doesn't address aging per se. |
| (173) | | RGs 1.83, Rev. 1 | Eddy-Current of Tubes. | Monitoring. | RG 1.83, Rev. 2, being developed from Steam Generator Integrity Program. | Better equipment and methods for defect detection, characterization, and sizing needed. | More accurate information needed on defect type and size to predict remaining tube integrity. | Rev.2 of RG 1.83 does not address performance demonstration issues. |
| (174) | | 1.85 | No explicit reference to aging; allows use of ASME approved and NRC code cases applicable to Section III. | The RG implies life extension by providing acceptance method for approved (ASME) materials acceptable to current Code Cases. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (175) | | 1.147 | No explicit reference to aging; allows use of ASME approved and NRC code cases applicable to Section XI. | The RG implies life extension by providing acceptance method for approved (ASME) inspection methods using current Code Cases. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (176) | | 1.158 | Qualification for inspection personnel. | Monitoring. | ASME Section XI Special Working Group developing performance demonstration requirements for ET inspections. | Better detection and sizing of defects needed especially for cracks. | Need methodology for qualifying ET inspectors. | |
| (177) | | ASME Section III, Subsection NB NE NF | Requirements-full-strength and pressure integrity. | None. | None. | N/A | New guide to cover limit of deterioration is needed. | Rules cover construction requirements but do not cover deterioration. |
| (178) | | Section XI | Material and component condition assessment. | Ref. Info./Data bases as they apply to requests for license extensions/renewals. | New Nonmandatory appendix on Record Keeping has passed the SWG-PLEX. | | Ref. info/data bases to support extensions/renewals. | As materials and components age, the predictive capabilities for physical and mechanical property changes must improve. Since during a plant's service life an increasing amount of information/data will be available from surveillance programs, better procedures to evaluate and use this information/data must be developed. |
| (179) | | | Surveillance. | New baseline examination requirement. | A new task group on Baseline Examination has been established. | | Evaluate a new baseline examination for license extension requests. | |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (180) | SURVEILLANCE (contd) | ASME (contd) Section XI | Allows for inspections beyond 40 years. | Changes to Inspection Plans A and B which are currently based on a 40-year operating life based on four intervals, the sum of which = 40. | SWG has implemented revisions to IWA-2400, which will delete the 40-year limit currently contained in Section XI. Also, reviewing need for more frequent inspections. | Extended surveillance beyond 40 years. | Surveillance beyond 40 years. | The frequency of inspection should depend on the degradation mechanisms and the condition of the component. |
| (181) | | | Monitoring and Testing. | Information/Data for trend curves. | Development of new exam techniques to detect age-related degradation and fatigue being considered. | Information and Data. | Monitoring/Testing techniques. | |
| (182) | | | Monitoring and Testing. | Information/Data for trend curves. | Considering hydrostatic testing trends where changes are needed for aging effects. | Information and Data. | Monitoring/Testing techniques. | |
| (183) | | | Information and Data for trend curve projections. | Records-Information/Data. | SWG is considering development of mandatory appendix for recordkeeping. | Information and Data for trend curves. | Records-Information/Data. | This Appendix would provide the utility owners with guidance as to the type of records which may be beneficial in supporting a license extension request. |
| (184) | | | Fatigue Failures. | Anticipate and avoid fatigue failures. | Fatigue monitoring technology developed and demonstrated. Several facilities are proceeding. | Experience and test data for high-cycle effects. | Regulatory acceptance of alternate approach to design basis. | a) Failures unlikely from cycles considered in design. Design approach is conservative. Component features included to preclude fatigue failure. Components have high tolerance to flaws.<br>b) Fatigue cracking has occurred during operations:<br>• vibration; rapid thermal cycling; pre-existing flaws<br>• other conditions not considered in design. |
| (185) | | IWB-2500 | None. | None. | Regulatory Guide 1.83. | Yes. | Issue new Regulatory Guide 1.83. | 3% inspection is not adequate and full length is needed (tube sheet to tube sheet). Current rules require sequential ISI, if flaw(s) are detected. i.e., if something is found, sample size is increased. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (186) | SURVEILLANCE (contd) | ASME (contd) IWB-3510 | Acceptance Standard. | Acceptance Standard. | None. | None. | Evaluation and Guide for allowable indications and deterioration limits should be determined. | |
| (187) | | IWB-3512 | Acceptance Standard. | Acceptance Standard. | None. | None. | | |
| (188) | | IWB-3516 | Acceptance Standard. | Acceptance Standard. | None. | None. | | |
| (189) | | IWB-3519 | Acceptance Standard. | Acceptance Standard. | None. | None. | | |
| (190) | | IWB-3520 | Acceptance Standard. | Acceptance Standard. | None. | None. | | |
| (191) | | IWB-3521 | Yes. | Yes. | | | Allowable indication for U-bend flaw is now 40% through-wall. Growth rate shows this is too high. | Amount of tubes plugged in a steam generator could give guidance for license renewal. |
| (192) | | IWB-2413, 2430 | None. | None. | | Yes. | | Should have increased examination program after detection of flaws: 3-1/2 years between examinations is now possible. |
| (193) | | IWB-3630 | None. | | Regulatory Guide 1.83. | | Yes, detection and sizing of SCC and IGA. | States that evaluation of cracks or IGA, etc., shall be performed by analysis acceptable to regulatory authority. |
| (194) | | Appendix IV | None. | None. | | | | Appendix IV rewrite in process.<br>This section of ASME (Section XI) may be transferred to Section V. Section XI will be replaced with a performance demonstration and qualification guideline. This is in the process of being written now but could take two years. |
| (195) | | GSIs 18 | ? | None. | None. | ? | ? | Steam-line break with consequential small LOCA. |
| (196) | | 29 | Visual examination is the primary method to detect degradation by acid corrosion or erosion/corrosion. | None. | None. | Need NDE for testing bolts and studs. Does the inspection frequency or fraction inspected need to be increased for extended life? | None. | Bolting Degradation or Failure in nuclear plants * Priority - HIGH. |

-- Steam Generator

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | GSIs (contd) | | | | | | |
| (197) | • | 35 | Loose part monitoring program. | None. | None. | ? | ? | Degradation of Internal Appurtenances in LWRs * Priority - LOW. |
| (198) | | 66 | Mitigate or reduce steam generator tube degradation and rupture. | None. | None. | ? | ? | Steam Generator Requirements * Four Items Priority - HIGH. |
| (199) | | 67 | Supplement Tube Inspections. | None. | None. | ? | ? | Steam Generator Staff Actions * Priority - LOW. |
| (200) | | A-3 | ? | None. | None. | ? | ? | Westinghouse Steam Generator Tube Integrity * See NUREG-0371. |
| (201) | | A-4 | ? | None. | None. | ? | ? | CE Steam Generator Tube Integrity * See NUREG-0371. |
| (202) | | A-5 | None. | ? | None. | ? | ? | B&W Steam Generator Tube Integrity * See NUREG-0371. |
| (203) | | A-12 | Lamellar tearing and low-fracture toughness of steam generator and reactor pump support materials. | None. | None. | ? | ? | Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports * Solution made available in October 1983 with publication of NUREG-0577. |
| (204) | | A-15 | Operation of LWR results in slow corrosion of interior metal surfaces of the primary coolant system. These are activated by neutron flux when circulating through the reactor and plate out. Radiation levels increase in the primary system, thus complicating routine inspection and maintenance. | None. | None. | Assess impact of increases in radiation levels on inspection and maintenance. | ? | Primary Coolant System Decontamination and Steam Generator Chemical Cleaning. |
| (205) | | USIA-47 | Define generic criteria for plant specific reviews for steam generator overfill transients in PWRs and reactor overfills in BWRs. | None. | None. | ? | ? | Safety Implications of Control Systems. |

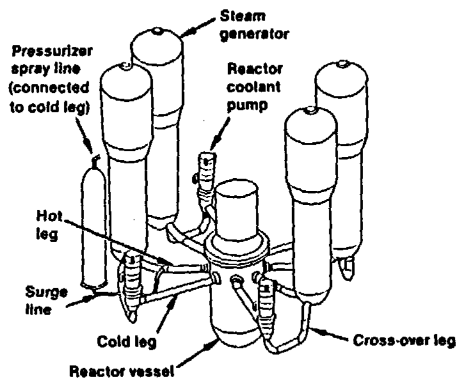
| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | GSIs (contd) | | | | | | |
| (206) | | B-60 | Presence of loose object in primary coolant system can be indicative of degraded reactor safety system resulting from failure or weakening component. | None. | None. | ? | Perform trend analysis on loose part monitoring data. | Loose Part Monitoring System * Priority - RESOLVED. |
| | | ANS | | | | | | |
| (207) | | 3.2 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | Quality Assurance documentation and/or records are required to show and evaluate deterioration of a component. |
| (208) | | 5.37 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |
| (209) | | 5.3.10 | None. | Review of plant procedures/records could give insight into deterioration. | None. | None. | None. | |

# APPENDIX IV

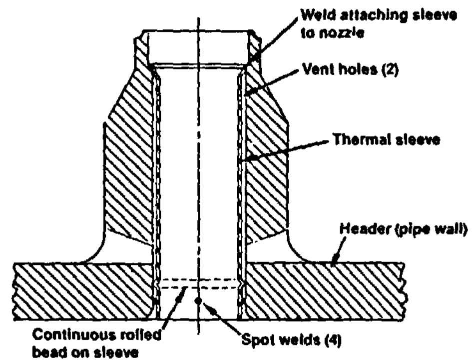
## <u>REGULATORY INSTRUMENT REVIEW FOR PRIMARY PIPING</u>

# Understanding and managing aging of PWR RCS piping and nozzles

| | | |
|---|---|---|
| Materials | Main coolant pipe | - Centrifugally cast SS-Gr. CF8A and CF8M (W), Type 304SS and 316SS (early W plants), SA-516 Gr. 70 (CE), SA-106 Gr. C (B&W) |
| | Fittings | - Statically cast SS - Gr. CF8A and CF8M (W); SA-516 Gr. 70, Type 309L SS (CE, B&W); Type 308L SS (B&W) |
| | Cladding | - Type 308L SS (CE, B&W)<br>Type 304L SS (B&W elbows) |
| | Surge and spray lines | - Type 316 SS, cast SS - Gr. CF8M (surge line in some CE plants) |
| | Charging, safety injection, and residual heat removal lines | - Type 316 SS |
| | Nozzles on main coolant pipe | - SA 105 Gr. 2 (CE), Type 304N SS (W) |
| | Safe ends | - Type 316 SS, Inconel SB-166 (CE, B&W) |
| | Thermal sleeves | - Inconel SB-166 |
| Stressors & Environment | Operational transients, temperature, flow induced vibrations, stratified flows, thermal striping, valve leakage, and thermal shocks | |



Typical RCS piping for W 4-loop plant



Typical nozzle-thermal sleeve assembly for a W plant

IV.1

| UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions) | | MANAGING AGING | | |
|---|---|---|---|---|
| **Sites** | **Aging Concerns** | **Inservice Inspections, Surveillance, and Monitoring** | | **Mitigation** |
| Nozzles and thermal sleeves<br>Charging<br>Safety injection<br>Surge<br>Spray | Low- and high-cycle thermal and mechanical fatigue | NRC requirements<br>Volumetric and surface examination of 25% of butt welds, including the following welds during each inspection interval (10 CFR 50.55a, IWB-2500):<br>- All terminal ends in each pipe connected to vessels<br>- All dissimilar metal welds<br>- All welds having cumulative usage factor equal to or greater than 0.4<br>- All welds having primary plus secondary stress intensity range equal to or greater than 2.4 $S_m$<br>Same welds are required to be inspected during each inspection interval<br>Flaw evaluation (10 CFR 50.55a, IWB-3000)<br>Leakage and hydrostatic pressure tests (10 CFR 50.55a, IWA-5000)<br>Cycle counting of specified design transients (Tech. Spec. requirement) | Recommendations<br>Perform more frequent examinations of nozzle welds and horizontal piping welds having significant fatigue damage<br>Determine fatigue damage by on-line monitoring of pipe wall temperatures and coolant flows, temperatures, and pressures in nozzles and horizontal portions of piping subject to operational transients, including stratified flows and thermal sleeves<br>Perform nondestructive examinations and loose parts monitoring to assess status of thermal sleeves<br>Develop use of acoustic emission method to detect crack growth in the base metal and welds<br>Develop techniques to monitor actual degree of thermal embrittlement in cast stainless steel piping:<br>- Analytical modelling of inservice degradation<br>- Metallurgical evaluation to characterize microstructure<br>- NDE to establish correlation between ultrasonic attenuation and fracture toughness<br>Monitor valve leakage in safety injection and residual heat removal lines<br>Develop UT to detect flaws in cast stainless steel piping | Maintain full flow in spray line and operate it continuously to prevent stratified flow and thermal shock conditions<br>Redesign surge and spray line piping by replacing short horizontal sections with sloped sections to prevent stratified flow conditions<br>Redesign piping to eliminate deleterious effects of valve leakage<br>Minimize valve leakage<br>Maintain smaller temperature differences between pressurizer and hot leg coolants during heat up and cool down<br>Maintain smaller temperature differences between the pressurizer and spray line coolants |
| Terminal end dissimilar metal welds (between carbon steel components and stainless steel piping) | Low- and high-cycle thermal and mechanical fatigue | | | |
| Surge and spray lines<br>Charging, safety injection, and residual heat removal lines to first isolation valve | Low- and high-cycle thermal and mechanical fatigue | | | |
| Cast stainless steel piping<br>Hot leg<br>Cross-over leg<br>Cold leg<br>Fittings<br>Surge line | Thermal embrittlement<br>- Coolant temperature<br>- Ferrite content and spacing<br>- Chemical composition of base metal | | | |

rev. 4

8-7429

# REGULATORY INSTRUMENTS REVIEW FOR PRIMARY PIPING

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| PRIMARY PIPING (1) | STRESS:<br>* STRESS CORROSION CRACKING IGSCC<br>* WELDMENTS<br>a. FERRITIC DISSIMILAR METALS OF SAFE ENDS<br>* FATIGUE<br>a. LOW CYCLE FATIGUE<br>b. THERMAL AND MECHANICAL<br>(SEE FIRST COMMENT OF COLUMN 9) | CFRs<br>10 CFR 50 | | | Nuplex codes and Standards work. Identify obvious changes, such as SSC in NUREG-0313. | Eliminate applicable unintentional embedded 40-year constraints in Regulatory Instrument codes and standards. | | Fatigue is generic to all types of plants, and causes and extent differ from plant to plant. It is also recognized that some unique aging problems do exist between PWRs and BWRs.<br><br>Forty-year constraints do not apply to all aging factors; i.e., fatigue is a cyclic phenomena and could become excessive after 5 or 10 years of operation. |
| (2) | | TSs<br>3/4.0.5 &<br>3/4.4.10 | None. | None. | None. | Depends on whether Section XI ISI requirements/reports are adequate vs. what needs more attention. | Depends on whether Section XI ISI requirements/reports are adequate vs. what needs more attention. | This TS requires ASME Section XI requirements concerning ISI and maintenance for RCS piping (IAW T.S. 3/4 4.0.5). |
| (3) | | 3/4.4.9.1.1 | Primary side thermal/pressure cycles are recorded during RCS heat-ups and cool-downs. | None, cycles associated with start-ups and shut-downs are recorded and counted. | None (see comment). | Cycles are recorded. Might be useful in evaluating thermal/pressure cycle history. | Effective methods to monitor fatigue cycles for magnitude and effect over the component's service life. | EPRI-sponsored fatigue monitoring "Fatigue-Pro" could impact the credibility of TS cycle count data. Fatigue-Pro is reported to be a measure of the severity of the cycles during operation, i.e., many operating cycles are less severe than assumed during the design. |
| (4) | | TSs All | | | Section XI SWG on Operating Plant Criteria is developing new non-mandatory appendixes to address return to service situations when TSs have been exceeded. | | | |
| (5) | | SRPs<br>3.9.1 | Meeting GDC 15: design with sufficient margin. Also, GDC 1,2,14; 10 CFR 50, Appendix B (QC): 10 CFR 100, Appendix A. | None (except for initial margins). | ? | None. | ? | Evaluate transients used in the design and fatigue analysis. |
| (6) | | 5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME Section III requirements for pressure boundaries). | None. | None. | None. | ? | Doesn't specifically address any one issue; ASME, Section III, must be followed. |

General Notes:
1. A "?" indicates further study/investigation is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessarily the aging issue.
3. For meaning of abbreviations, acronyms, and initialisms, used throughout, see acronyms on page xi, xii, and xiii of the report.

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | STRESS (contd) | SRPs (contd) | | | | | | |
| (7) | | 5.2.3 | Review suitability of materials chosen. Perform tests (fracture toughness). Review procedures of manufacture welding. NDE Section III NB-2500, NB-2550, and NB-2570. | None. | ? | None. | ? | Covers all Reactor Coolant Pressure Boundary Materials |
| | | RGs | | | | | | |
| (8) | | 1.147 | No explicit reference to aging; provides an NRC acceptable list of approved ASME code cases for Section XI. | The RG implies life extension by providing acceptance method for approved (ASME) inspection methods using current Code Cases. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (9) | | 1.85 | No explicit reference to aging; provides an NRC acceptable list of approved ASME code cases for Section III. | The RG implies life extension by providing acceptance method for approved (ASME) Code Case materials and testing acceptable to NRC staff. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| | | ASME | | | | | | |
| (10) | | Section III | | Requalification of components. | A new appendix is under consideration which may be used to requalify components which have exceeded the rules for cyclic operation in Section III, NB-3222.4. A task group has been formed to address the issue. This also applies to Section XI. | Evaluate new appendix. | Requalification Rules for components. | |
| (11) | | | Fatigue. | Fatigue curve revisions. | Section XI, SWG on Operating Plant Criteria is reviewing fatigue curves in Section III to determine if they can be revised to accommodate operation beyond 40 years. | | Improved end-of-life projections, 40 years and beyond. | Studies are under way to assess the magnitude of the effects of actual environmental conditions. Code curves are based on smooth specimens in air at room temperature, whereas materials in service have much rougher surfaces and are exposed to flowing coolant at operating temperatures. The actual surfaces are more prone to crack initiation. Studies and research to assess magnitude of effect of environmental factors could result in need for new Regulatory Guide, Regulatory Instruments and/or code revisions. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | STRESS (contd) | ASME (contd) | | | | | | |
| (12) | | Subsection NB NC | Provides for design loads affecting the strength and integrity of the pressure boundary. | None-explicit; provides rules for repair and/or replacement of code components. | | Implied in the design criteria (see comments). | Guidance is needed to cover the limits of component deterioration. | ASME Code for Class 1 components-piping design required when corrosion or erosion is expected; the piping thickness shall be increased over the other design requirements. This allowance shall be consistent with the specified design life.<br><br>Rules cover design and construction requirements but do not cover deterioration. |
| (13) | | Section XI | Condition Assessment. | Requalification of systems and components. | Consideration of development of new "Article IWX-8000, Requalification for Fatigue Life." | Condition Assessment information/data. | Requalification of systems and components following expiration of operating license. | |
| (14) | | | TSs have been exceeded. | | SWG on Operating Plant Criteria is developing new non-mandatory appendixes to address return to service situations, when TSs have been exceeded. | | | |
| (15) | | GSIs 14 | Cracking in high pressure piping in PWRs as a result of corrosion, vibratory and thermal fatigue, and dynamic loading. | None. | None. | None. | None. | PWR pipe cracks<br>* Priority - LOW, RESOLVED.<br>* Corrosion cracking in low-pressure piping is addressed in C-7. |
| (16) | | 73 | Fatigue failure problems connected with nozzle-thermal sleeve assemblies. | None. | ? | ? | ? | Detached thermal sleeves - The thermal sleeves were installed in the safety injection accumulator piping nozzle connections to the reactor coolant system cold legs piping. Found at bottom of reactor vessels. |
| (17) | | 86 | IGSCC caused leaks in heat-affected zones of the safe-end-to-pipe welds. | None. | ? | ? | ? | Long range plan for dealing with stress corrosion cracking in BWR piping - solution available. |
| (18) | (Not used) | | | | | | | |
| (19) | | A-10 | BWR feedwater cracking due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. | None. | None. | BWR feedwater nozzle cracking solutions and/or improved repair methods. | | |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | STRESS (contd) | GSIs (contd) | | | | | | |
| (20) | | A-15 | Operation of LWR results in slow corrosion of interior metal surfaces of the primary coolant system, then these products are activated by neutron flux and when circulated through the reactor plate-out occurs. | None. | None. | ? | ? | Primary coolant system decontamination and steam generator chemical cleaning. |
| (21) | | A-42 | Pipe cracking due to IGSCC of austenitic stainless steel components made susceptible to this by being sensitized by post-weld heat treatment or by sensitization of a narrow heat-affected zone near welds. | None. | None. | More effective NDE to assure absence of IGSCC, especially under weld-overlay repaired areas. | Evaluate need to replace susceptible and/or weld-overlay repaired piping. | Pipe cracks in BWRs<br>* Priority - RESOLVED. |
| (22) | | B-6 | Through-wall cracks and fatigue crack growth. | None. | None. | ? | ? | Loads, load combinations, and stress limits.<br>* Priority - HIGH.<br>* See NUREG-2800 Supl. |
| (23) | | B-25 | ? | None. | None. | ? | ? | Piping Benchmark Problems. |
| (24) | | C-7 | Combinations of fabrication, stress and environmental conditions have resulted in isolated instances of stress corrosion cracking of low-pressure 304 piping. | None. | None. | ? | ? | PWR System Piping<br>* Priority - RESOLVED. |
| (25) | DYNAMIC EFFECTS<br>* VIBRATION<br>* THERMAL CYCLES<br>a. THERMAL AGING<br>b. THERMAL SHOCK<br>* EROSION<br>a. COMBINATION--CORROSION/EROSION | CFRs<br>10 CFR 50, Appendix G, I, II and IV | ISI programs per ASME Code requirements. | None. | None. | ISI coverage may need to be increased. | ISI criteria may be different for License Renewal. | ISI research needs should be identified. Assisted effects (Note: exposure rates in ex-vessel components may be too low to cause an effect). |

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|---|---|---|---|---|---|---|---|---|
| (26) | DYNAMIC EFFECTS (contd) | TSs<br>3/4.0.5 &<br>3/4.4.10 | None. | None. | None. | Depends on whether Section XI ISI requirements/reports are adequate vs. what needs more attention. | Depends on whether Section XI ISI requirements/reports are adequate vs. what needs more attention. | This TS requires ASME, Section XI, requirements concerning ISI and maintenance for all RCS piping that is code class 1, 2, or 3 (IAW T.S. 3/40.5). |
| (27) | | 3/4.4.9.1.1 | RCS thermal cycles are recorded during heat-ups and cool-downs. | None. | None. | Cycles are recorded. Might be useful in the evaluation of thermal cycle history. | May need to evaluate thermal cycle history. | |
| (28) | | 3.9.2 | Review the criteria, testing procedures, and dynamic analysis. | None. | ? | None. | ? | Assure structural and functional integrity under vibratory loadings. |
| (29) | | 5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME, Section III, requirements). | None. | None. | None. | ? | Doesn't address specific issues; ASME Section III must be followed. |
| (30) | | RGs<br>1.147 | No explicit reference to aging; provides an NRC acceptable list of ASME approved code cases applicable to Section XI. | The RG implies life extension by providing acceptance method for approved (ASME) inspection methods using current Code Cases. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (31) | | 1.85 | No explicit reference to aging; provides an NRC acceptable list of ASME approved code cases applicable to Section III. | The RG implies life extension by providing acceptance method for approved (ASME) Code Case materials and testing acceptable to NRC Staff. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases re usually superseded (annulled) by revisions the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (32) | | ASME<br>Section III<br>Subsection<br>NB<br>NC | Provides for design requirements for impact loads, earthquake, vibrations, RV thrusts, weights and the effects of thermal expansion and contration. | None explicit; provides rules for repair and/or replacement of code components. | | Implied in the design criteria. | Guidance is needed to cover the limits of component deterioration. | ASME Code for Class I components-piping design required when corrosion or erosion is expected; the piping thickness shall be increased over the other design requirements. This allowance shall be consistent with the specified design life.<br><br>Rules cover design and construction requirements but do not cover deterioration. |

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|---|---|---|---|---|---|---|---|---|
| (33) | DYNAMIC EFFECTS (contd) | ASME (contd) Section XI | Corrosion and Erosion. | ? | A new SWG on pipe wall-thinning has been established and will review this issue with EPRI. Main area of concern at this time is the effect of erosion/corrosion on piping in non-nuclear applications (ANSI/ASME B-31G-1984), the Remaining Strength of Corroded Pipelines. | ? | ? | The SWG principally applies to secondary systems, not primary.<br><br>The Section XI Subgroup on Nondestructive Examination is currently developing a Code Case and non-mandatory appendix on UT Detection and Measurement of Erosion/Corrosion.<br><br>ANSI/ASME B-31G-1984, "Manual for Determining the Remaining Strength of Corroded Pipelines," addresses non-nuclear applications, but may have application to nuclear components. |
| (34) | (Not used) | | | | | | | |
| (35) | | GSIs 14 | Cracking in high pressure piping in PWRs as a result of corrosion, fatigue, and dynamic loading. | None. | None. | None. | None. | PWR pipe cracks<br>* Priority - LOW, RESOLVED.<br>* Corrosion cracking in low-pressure piping is addressed in C-7. |
| (36) | | 73 | Fatigue failure problems connected with nozzle-thermal sleeve assemblies. | None. | None. | ? | ? | Detached thermal sleeves<br><br>* Thermal sleeves installed in the safety injection accumulator piping nozzle connections, to the reactor coolant system cold legs piping, were found at the bottom of reactor vessels. |
| (37) | | 86 | IGSCC-caused leaks in heat-affected zones of the safe-end-to-pipe welds. | None. | None. | ? | ? | Long-range plan for dealing with stress corrosion cracking in BWR piping<br>* Solution available. |
| (38) | (Not used) | | | | | | | |
| (39) | | A-1 | Water hammer incidents involving steam generator feed rings and piping, emergency core cooling systems, RHR systems, containment spray, service water feedwater and steam lines. | None. | None. | ? | ? | Water hammer<br>* RESOLVED 3/15/84 with publication of NUREG-0927. (Complete resolution of all water hammer problems remains as a debatable question.) |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (40) | DYNAMIC EFFECTS (contd) | GSIs (contd) A-10 | BWR feedwater cracking due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. | None. | None. | ? | ? | BWR feedwater nozzle cracking. |
| (41) | | A-15 | Operation of LWR results in slow corrosion of interior metal surfaces of the primary coolant system, then these products are activated by neutron flux and when circulated through the reactor plate-out occurs. | None. | None. | ? | ? | Primary coolant system decontamination and steam generator chemical cleaning. |
| (42) | (Not used) | | | | | | | |
| (43) | (Not used) | | | | | | | |
| (44) | (Not used) | | | | | | | |
| (45) | (Not used) | | | | | | | |
| (46) | STRUCTURAL INTEGRITY * FRACTURE • TOUGHNESS * EMBRITTLEMENT • THERMAL * SEISMIC (DAMAGE/FAILURES • HANGER, SNUBBERS AND ANCHORS | CFRs 10 CFR 50, Appendix A, General Design Criteria (GDC) | Provides principal design criteria for the testing and performance requirements for components. | None. | None. | N/A, doesn't change over time. | Does the existing design (generic or specific), meet the current (present time) acceptable testing and performance standards for life extension? | |
| (47) | | Appendix A, | Design of components should be accomplished with consideration for natural phenomena, such as earthquakes, tornadoes, and floods. | Design is based on historical data from the time of original license; accuracy and information may have been altered. | None. | Requirements may change over time. "Natural" events may alter the design base. | Analyses of current known conditions as applicable will be required. | Generic or site specific research may be Criterion 2 necessary. |
| (48) | | TSs 3/4.0.5 & 3/4.10 | None. | None. | None. | Depends on whether ASME, Section XI ISI, requirements are adequate vs. what needs more attention. | Depends on whether ASME, Section XI ISI, requirements are adequate vs. what needs more attention. | This TS requires the requirements of T.S. 4.0.5. (ASME, Section XI; ISI) be applied to RCS piping. |

-- Primary Piping

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (49) | STRUCTURAL INTEGRITY (contd) | TSs (contd) 3/4.7.9 | Requires periodic testing of snubbers. | None. | None. | Might need to expand to include dead weight supports, anchors, etc., as necessary. Inspections are to be the basis for hanger evaluation. | Could be modified to ensure records are adequate for LR assessment. | This TS requires periodic functional testing of snubbers. |
| (50) | | TSs | | | Working Group on Operating Plant Criteria; a proposed addition to Article IWB-3000 of Section XI would contain rules and formulas for performing an evaluation of a component's fitness for continued service when either TSs or elements of its construction/design limitations had been exceeded. | | | |
| (51) | | SRPs 3.6.2 | Review original system design adequacy system. | None. | None. | None. | ? | Mainly concerned with the consequences of a pipe break (pipe whip and jet forces). By knowing where and how a pipe is most likely to break, implies providing adequate design margins to prevent such occurrences (i.e., the piping system is analyzed thoroughly). |
| (52) | | BTP MEB 3-1 | Stress and fatigue limits; design requirements; inspection of welds. | None. | ? | None. | ? | Breaks are usually at points of high stress and fatigue (terminal ends and nozzles). |
| (53) | | 3.9.3 and 3.9.3, Appendix A | Reference to ASME Section III and GDC 1,2,4,14,15. | None. (except for original design margins). | ? | None. | ? | Loading combinations, system operational transients, and stress limits. |
| (54) | | 5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME, Section III, requirements for pressure boundaries). | None. | None. | None. | ? | |
| (55) | | 5.4.3 | Meeting requirements of SRP, Section 3.9.1, 3.9.2, 3.9.3, 5.2.3, and 5.2.4. | None. | None. | None. | ? | |
| (56) | | RGs 1.31, Rev 3 | Control of ferrite in stainless steel welds to limit microcracking. | None. | None. | Define level of ferrite needed to resist environmentally assisted cracking. | Determine life expectancy of weld overlay repairs. | Intent of RG 1.31 wasn't to specify ferrite level needed to resist cracking. Results from EPRI RP T302-2 suggest currently accepted ferrite levels may be too low. |

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|---|---|---|---|---|---|---|---|---|
| (57) | STRUCTURAL INTEGRITY (contd) | RGs (contd) 1.46 | Protection against pipe whip inside containment. | Withdrawn 3/1/85. See SRP 3.9.2. | None. | | | Cast Stainless Steel is a generic issue in primary piping systems, i.e., flanges, pump casing, etc. |
| (58) | | 1.147 | No explicit reference to aging; provides an NRC acceptable list of ASME approved code cases applicable to Section XI. | The RG implies life extension by providing acceptance method for approved (ASME) inspection methods using current Code Cases. | RG revisions are made include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (59) | | 1.48 | Design and loading combinations for seismic Category I, fluid system components. | Withdrawn 3/1/85. See SRP 3.9.3 | None. | | | |
| (60) | | 1.85 | No explicit reference to aging; provides an NRC acceptable list of ASME approved code cases applicable to Section III. | The RG implies life extension by providing acceptance method for approved (ASME) Code Case materials and testing acceptable to NRC staff. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |
| (61) | | ASME Section III Subsection NB NC | Provides for design requirements for impact loads, earthquake, vibrations, RV thrusts, weights and the effects of thermal expansion and contraction. | None explicit; provides rules for repair and/or replacement of code components. | | Implied in the design criteria. | Guidance is needed to cover the limits of component deterioration. | ASME Code for Class I components-piping design, requires when corrosion or erosion is expected the piping thickness shall be increased over the other design requirements. This allowance shall be consistent with the specified design life. Rules cover design and construction requirements but do not cover deterioration. |
| (62) | | GSIs 14 | Cracking in high pressure piping in PWRs as a result of corrosion, vibratory and thermal fatigue, and dynamic loading. | None. | None. | None. | None. | PWR pipe cracks<br>* Priority - LOW, RESOLVED.<br>* Corrosion cracking in low pressure piping is addressed in C-7. |
| (63) | | 86 | | | | | | Long-range plan for dealing with stress corrosion cracking in BWR piping. |
| (64) | | 119 | None. | None. | None. | None. | None. | Piping review committee recommendations - "No significant change in public safety will result from resolution of this issue." |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | STRUCTURAL INTEGRITY (contd) | GSIs (contd) | | | | | | |
| (65) | | A-1 | Water hammer incidents involving steam generator feed rings and piping, emergency core cooling systems, RHR systems, containment spray, service water feedwater and steam lines. | None. | None. | ? | ? | Water hammer<br>* RESOLVED 3/15/84 with publication of NUREG-0927. |
| (66) | | A-10 | BWR feedwater cracking due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. | None. | None. | ? | ? | BWR feedwater nozzle cracking. |
| (67) | | A-15 | Operation of LWR results in slow corrosion of interior metal surfaces of the primary coolant system, then these products are activated by neutron flux and when circulated through the reactor plate-out occurs. | None. | None. | ? | ? | Primary coolant system decontamination and steam generator chemical cleaning. |
| (68) | | A-42 | None. | None. | None. | None. | None. | Pipe cracks in BWRs<br>* Priority - RESOLVED. |
| (69) | | B-6 | None. | None. | None. | None. | None. | Loads, load combinations, and stress limits<br>* Priority - HIGH. |
| (70) | | B-25 | ? | None. | ? | ? | ? | Piping Benchmark Problems. |
| (71) | | C-7 | None. | None. | None. | ? | ? | PWR System Piping<br>* Priority - RESOLVED. |

-- Primary Piping

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (72) | CORROSION:<br>* WATER CHEMISTRY<br>* IMPURITIES<br>* INTERGRANULAR ATTACK (IGA)<br>* COMBINATION--CORROSION-EROSION<br>* IRRADIATION ASSISTED 5.2.3 EFFECTS (INCREASED RADIO-LYTIC DECOMPOSITION OF WATER) | CFRs<br>10 CFR 50, Appendix A, General Design Criteria (GDC) | Provides principal design criteria for component testing and performance requirements. | None. | None. | N/A, doesn't change over time. | Does the existing design (generic or specific), meet currently acceptable testing and performance standards for life extension? | |
| (73) | | TSs<br>3/4.0.5 & 3/4.4.10 | Requires ASME, Section XI, ISI for code class 1, 2, and 3 components. | None. | None. | N/A | Depends on whether reports/records of ASME, Section XI, are adequate to be useful. | This TS requires the requirements of T.S. 3/4.0.5 (ASME, Section XI; ISI) to be applied to code class 1, 2, 3 piping. (Note: class 2 & 3 are generally not primary systems.) |
| (74) | | 3/4.4.4.7 | Requires chemistry control of the primary (RCS) coolant. | None. | None. | N/A | RCS water chemistry records are retained. Perhaps useful in evaluating piping history with regard to (RCS). | RCS chemistry limits are shown below. Are these adequate for the corrosion issue?<br>Parameter / Steady State Limit / Transient Limit<br>(dissolved $O_2$)* / <0.10 ppm / <1.00 ppm<br>($Cl^-$)* / <0.15 ppm / <1.50 ppm<br>($F^-$)* / <0.15 ppm / <1.50 ppm<br>*N/A if RCS temp. is <250°F. |
| (75) | | SRPs<br>5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME Section III requirements for pressure boundaries). | None. | None. | None. | ? | Doesn't address specific issues; ASME Section III must be followed. |
| (76) | | 5.2.3 | Review suitability of materials chosen. NDE testing per ASME Section III, NB-2500. | None. | ? | None. | ? | Covers all reactor coolant pressure boundary materials. |
| (77) | | RGs<br>1.44 | Process controls to minimize sensitization in stainless steel welds. | None. | None. | Better understanding of relationship between welding process and sensitization. | Determine life expectancy of sensitized material and nuclear-grade stainless steel. | NRC Program on "Evaluation of Welded and Repair-Welded Stainless Steel for LWR Service" initiated work in this area but was not completed. A small effort may be ongoing. |

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|---|---|---|---|---|---|---|---|---|
| | CORROSION (contd) | RGs (contd) | | | | | | |
| (78) | | 1.56, Rev. 1 | Control of water purity in BWRs. | None. | ANL programs on BWR pipe crack remedies. | Better knowledge of affect of impurities on cracking. | Life expectancy of cracked pipes on presence of impurities. | RG 1.56 limits only conductivity, pH and chloride content of water. Other species need to be considered. Recent research suggests limits may not be low enough. NUREG-0313 includes guidelines for BWR piping (IGSCC mitigation). |
| (79) | | ASME Section III Subsection NB NC | Provides for design loads affecting the strength and integrity of the pressure boundary. | None explicit; provides rules for repair and/or replacement of code components. | | Implied in the design criteria (see comments). | Guidance is needed to cover the limits of component deterioration. | ASME Code for Class 1 components piping design required when corrosion or erosion is expected; the piping thickness shall be increased over the other design requirements. This allowance shall be consistent with the specified design life.<br><br>Rules cover design and construction requirements but do not cover deterioration. |
| (80) | | GSIs 14 | Cracking in high pressure piping in PWRs as a result of corrosion, vibratory and thermal fatigue, and dynamic loading. | None. | None. | None. | None. | PWR pipe cracks<br>* Priority - LOW, RESOLVED.<br>* Corrosion cracking in low-pressure piping is addressed in C-7. |
| (81) | | 73 | Fatigue failure problems connected with nozzle-thermal sleeve assemblies. | None. | None. | ? | ? | Detached thermal sleeves<br>* Priority -<br>* The thermal sleeves installed in the safety injection accumulator piping nozzle connections to the cold leg piping were found at the bottom of reactor vessels. |
| (82) | | 86 | IGSCC-caused leaks in heat-affected zones of the safe-end-to-pipe welds. | None. | None. | ? | ? | Long-range plan for dealing with stress corrosion cracking in BWR piping.<br>* Solution available. |
| (83) | | 111 | Stress corrosion cracking in steam generator. | None. | None. | ? | ? | Stress corrosion cracking of pressure boundary ferritic steels in selected environments. |
| (84) | | 119 | None. | None. | None. | None. | None. | Piping review committee recommendations - "No significant change in public safety will result from resolution of this issue." |
| (85) | | A-1 | None. | None. | None. | None. | None. | Water hammer. |
| (86) | | A-10 | None. | None. | None. | None. | None. | BWR feedwater nozzle cracking. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | CORROSION (contd) | GSIs (contd) | | | | | | |
| (87) | | A-15 | Operation of LWR results in slow corrosion of interior metal surfaces of the primary coolant system, then these products are activated by neutron flux and when circulated through the reactor plate-out occurs. | None. | None. | ? | ? | Primary coolant system decontamination and steam generator chemical cleaning. |
| (88) | | A-42 | None. | None. | None. | None. | None. | Pipe cracks in BWRs<br>* Priority - RESOLVED. |
| (89) | | B-6 | None. | None. | None. | None. | None. | Loads, load combinations, and stress limits<br>* Priority - HIGH. |
| (90) | | B-25 | ? | None. | None. | ? | ? | Piping Benchmark Problems |
| (91) | | C-7 | None. | None. | None. | ? | ? | PWR System Piping<br>* Priority - RESOLVED |
| (92) | SURVEILLANCE:<br>* TESTING<br>* INSPECTION<br>* MAINTENANCE AND REPAIRS<br>* NDE | CFRs<br>10 CFR 50, Appendix A, General Design Criteria (GDC) | Provides principal design criteria for the testing and performance requirements for components. | None. | None. | N/A, doesn't change over time. | Does the existing design (generic or specific) meet currently acceptable testing and performance standards for life extension? | |
| (93) | | Appendix A, Criterion 2 | Design of components should be accomplished with consideration for natural phenomena, such as earthquakes, tornadoes and floods. | Design is based on historically data from the time of original license; accuracy and information may have been altered. | None. | Requirements may change over time. "Natural" events may alter the design base. | Analyses of current known conditions as applicable will be required. | Generic or site specific research may be necessary. |
| (94) | | TSs<br>3/4.0.5 & 3/4.10 | Requires ASME, Section XI, treatment of piping with regard to maintenance and ISI if code class 1, 2, or 3. | None. | None. | N/A | ? | TS require the provisions of T.S. 3/4.0.5 (ASME, Section XI, Inservice Inspection) be applied to Code Class 1, 2 & 3 Piping. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (95) | SURVEILLANCE (contd) | TSs (contd) | | | NSPE-12-4311 NUPLEX report to be prepared on reactor coolant pressure boundary piping (see comments). | | | Description: Perform wall thickness trending on primary/secondary piping systems. Also requirements for transient monitoring of primary piping high-stress locations and thermal duty. Establish piping vibration and dynamic effects measurement piping/sliding support locations for wear/corrosion.<br><br>Could lead to TSs modifications of NRC rule changes or codes and Standards. |
| (96) | | SRPs<br>5.2.1.1 | Must meet 10 CFR 50.55a (i.e., meet ASME, Section III, requirements for pressure boundaries). | None. | None. | None. | ? | Doesn't address specific issues; ASME Section III must be followed. |
| (97) | | 5.2.4 | Require ISI Program of RCPB (GDC-32) to assess the structural and leak-tight integrity. | None. | ? | ? | ? | Based on requirements of 10 CFR 50.55a and detailed in ASME, Section XI. |
| (98) | | 17.2 | Implement QA program. | None. | ? | None. | ? | Doesn't address aging per se; just states that a QA program is required. |
| (99) | | RGs<br>1.11 | Visual inspection of instrument lines. | Monitoring. | None. | None. | None. | |
| (100) | | 1.45 | Systems for detection of leakage in the RCPB. | None. | ANL development of acoustic leak monitoring. | Correlate leak rate with crack size to define acceptance limits. | None. | ANL has developed an acoustic leak monitoring system from laboratory experiments. The system needs field validation. |
| (101) | | 1.58, Rev 1 | Qualification for inspection personnel. | Monitoring. | ASME Section XI developing criteria for performance demonstration qualification of inspectors. | Better detection and sizing of crack-type defects. | Better training and qualification procedures. Improved inspection methods for coarse-grain materials (welds, cast SS, etc.). | |
| (102) | | 1.147 | No explicit reference to aging; provides an NRC acceptable list of ASME approved code cases applicable to Section XI. | The RG implies life extension by providing acceptance method for approved (ASME) inspection methods using current Code Cases. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. Code Cases provide a mechanism to use alternate methods within jurisdiction of the code. The cases are usually superseded (annulled) by revisions to the code, i.e., the case becomes part of the code. In other instances, the case may be reversed or dropped as an applicable method. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (103) | SURVEILLANCE (contd) | RGs (contd) NUREG-1061 | | | NSPE-12-4240 NUPLEX report to be prepared on reactor recirculation lines (see comments). | | | Description: Increase recirculation piping inspection intervals and percentages to comply with NUREG-1061. Develop crack growth curves for austenitic ss in NWC environment, considering IHSI and other weld treatments. Develop inspection/monitoring program for thermal embrittlement of duplex cast SS pump and valve bodies. |
| (104) | | ASME Section III NB-3600 | | | Code requirements updated to include transient considerations startup testing for vibration and thermal displacements required. | | | Early plants designed to simplified criteria of ASME VIII and B31.1. |
| (105) | | Section XI | Material and component condition assessment. | Ref. info./data bases as they apply to requests for licence extensions/renewals. | New nonmandatory appendix on recordkeeping has passed SWG-PLEX. | See other columns. | Ref. info/data bases to support extensions/renewals. | As materials and components age, the predictive capabilities for physical and mechanical property changes must improve. Since, during a plant's service life, an increasing amount of information/data will be available from surveillance programs, better procedures to evaluate and use this information/data can and must be developed. |
| (106) | | | Surveillance. | New baseline examination requirements. | A new task group on baseline examination has been established. | See other columns. | Possible need to require a new baseline examination for a license extension request. | |
| (107) | | | Allows for inspections beyond 40 years. | Changes to inspection Plans A and B which are currently based on a 40-year operating life based on four intervals, the sum of which = 40. | SWG has implemented revisions to IWA-2400, which will delete the 40-year limit currently contained in Section XI. Also reviewing need for more frequent and extensive inspections. | Extended and/or enhanced surveillance beyond 40 years. | Surveillance beyond 40 years. | |
| (108) | | | Monitoring and testing. | Information/data for trend curves. | Development of new exam. techniques to detect age-related degradation and fatigue being considered. | Information, data, and assessments. | Monitoring/testing techniques. | |
| (109) | | | Information and data for trend projections. | Records, information/data. | A SWG is considering development of a nonmandatory appendix for recordkeeping. | Information and data for trend curves. | Records, information/data. | This appendix would provide the utility with guidance as to the records needed to support a license extension request. |

-- Primary Piping

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | ASME (contd) | | | | | | |
| (110) | | | Fatigue failures. | Anticipate and avoid fatigue failures. | Fatigue monitoring technology developed and demonstrated; several utilities proceeding. | Experience and test data for high-cycle effects. | Regulatory acceptance of alternate approach to design bases. | a) Failures unlikely from cycles considered in design. Design approach is conservative. Component features included to preclude fatigue failure. Components have high tolerance to flaws.<br>b) Fatigue cracking has occurred during operations:<br>• vibration<br>• rapid thermal cycling<br>• pre-existing flaws<br>• other conditions not considered in design. |
| | | GSIs | | | | | | |
| (111) | | 14 | Cracking in high pressure piping in PWRs as a result of corrosion, vibratory and thermal fatigue, and dynamic loading. | None. | None. | None. | None. | PWR pipe cracks<br>* Priority - LOW RESOLVED.<br>* Corrosion cracking in low-pressure piping is addressed in C-7. |
| (112) | | 73 | Fatigue failure problems connected with nozzle-thermal sleeve assemblies. | None. | None. | ? | ? | Detached thermal sleeves - The thermal sleeves installed in the safety injection accumulator piping nozzle connections to the cold leg piping were found at the bottom of reactor vessels. |
| (113) | | 86 | IGSCC-caused leaks in heat-affected zones of the safe-end-to-pipe welds. | None. | None. | ? | ? | Long-range plan for dealing with stress corrosion cracking in BWR piping - solution available. |
| (114) | | 111 | Stress corrosion cracking in steam generator. | None. | None. | ? | ? | Stress corrosion cracking of pressure boundary ferritic steels in selected environments. |
| (115) | | 119 | None. | None. | None. | None. | None. | Piping review committee recommendations - "No significant change in public safety will result from resolution of this issue." |
| (116) | | A-1 | None. | None. | None. | None. | None. | Water hammer. |
| (117) | | A-10 | None. | None. | None. | None. | None. | BWR feedwater nozzle cracking. |

-- Primary Piping

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | GSIs (contd) | | | | | | |
| (118) | | A-15 | Operation of LWR results in slow corrosion of interior metal surfaces of the primary coolant system, then these products are activated by neutron flux and when circulated through the reactor plate-out occurs. | None. | None. | ? | ? | Primary coolant system decontamination and steam generator chemical cleaning. |
| (119) | | A-42 | None. | None. | None. | None. | None. | Pipe cracks in BWRs<br>* Priority - RESOLVED. |
| (120) | | B-6 | None. | None. | None. | None. | None. | Loads, load combinations, and stress limits<br>* Priority - HIGH. |
| (121) | | B-25 | ? | None. | None. | ? | ? | Piping Benchmark Problems. |
| (122) | | C-7 | None. | None. | None. | ? | ? | PWR System Piping<br>* Priority - RESOLVED. |

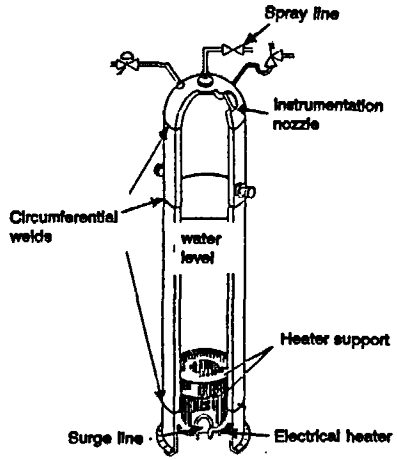
-- Primary Piping

# APPENDIX V

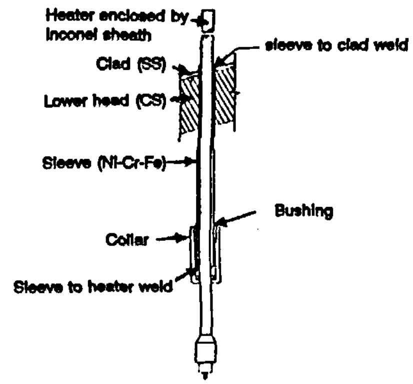
## REGULATORY INSTRUMENT FOR PRESSURIZER (INTERNALS AND SUPPORT PIPING)

## Understanding and managing aging of pressurizer, surge & spray lines

| | | |
|---|---|---|
| Materials | Vessel | • shell, A-533, Gl. B, Class I.<br>Cladding, Type 304 SS & Ni-Cr-Fe Alloy |
| | Heater | • Sheath, Inconel Insulation MgO |
| | Fittings | • Statically cast SS - Gr. CF8A and CF8M (W); SA 516 Gr. 70, Type 309L SS (CE, B&W); Type 308L SS (B&W) |
| | Cladding | • Type 308L SS (CE), Type 304L SS (B&W) |
| | Surge line | • Type 316 SS, cast SS-Gr. CF8M some (CE plants) |
| | Spray line<br>Nozzles on main coolant pipe<br>Thermal sleeve | • Type 316 SS<br>• SA 105 Gr. 2 (CE), Type 304 N SS (W)<br>• Inconel SB-168 |
| Stressors and Environment | | Operational transients, temperature, flow induced vibrations, stratified flows, thermal stripping, thermal shocks, heater mechanical wear and element burnout and erosion and corrosion |




Typical W pressurizer and connections




Typical CE heater equipment

| UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions) | | MANAGING AGING | | |
|---|---|---|---|---|
| **Sites** | **Aging Concerns** | **Inservice Inspections, Surveillance, and Monitoring** | | **Mitigation** |
| Nozzles and thermal sleeves<br>• Instrumentation<br>• Surge<br>• Spray | Low and high-cycle thermal<br>Erosion | <u>NRC requirements</u><br>Volumetric and surface examination of 25% of butt welds including the following welds each inspection interval 10 CFR 50.55a, IWB-2500:<br>• All dissimilar metal welds<br>• All welds having cumulative usage factor equal to or greater than 0.4<br>• All welds having stress intensity range of 2.4 $S_m$<br>Same welds are required to be inspected during each inspection interval<br>Flaw detection and evaluation 10 CFR 50.55a, IWB-3000<br>Leakage Hydrostatic pressure tests 10 CFR 50.55a, IWB-5000<br>ASME Section III, NB-3210 and ASME Section XI, ISI<br>Cycle counting of specified design transients<br>Tech Spec's requirements<br>• Cycle counting of specified design transients<br>• Leakage rates<br>• $\Delta$T limits for heatup/cooldown | <u>Recommendations</u><br>Perform more frequent examination of nozzle welds having high cumulative usage factor.<br>Determine fatigue damage by on-line monitoring of coolant and piping temperatures, and flow rates in nozzles and horizontal portions of piping during operational transients, stratified flows, and thermal shocks.<br>Perform nondestructive examinations and loose parts monitoring to assess status of thermal sleeves develop improved NDE method to detect crack growth in the base metal and welds<br>Develop techniques to monitor actual degree of thermal embrittlement, e.g., develop improved NDE methods and tools using magnetic properties measurements and acoustic emission<br>Monitor valve leakage<br>Develop UT to detect flaws on cast stainless steel piping | Maintain full flow in spray line and operate it continuously to prevent stratified flow and thermal shock conditions<br>Replace horizontal section of spray line with sloped section to prevent stratified flow condition<br>Redesign piping to eliminate valve leakage<br>Preventitive or predictive maintenance for heater replacement<br>Use improved stem packing materials |
| Terminal end dissimilar metal welds (between carbon steel components and stainless steel piping | Low-Cycle thermal and mechanical fatigue | | | |
| Cast stainless steel piping<br>• Surge line<br>• Spray line<br>• Valves<br>• Fittings | Low-cycle thermal and mechanical fatigue<br>Thermal embrittlement<br>Erosion spray valve<br>Boric acid Corrosion<br>Stem packing/wear and degradation with age and service life<br>Bellow degradation | | | |
| Heater sheath failures | Small LOCA via heater element and heater sleeve | | | |
| Vessel wall | High and low cycle thermal fatigue | | | |

## REGULATORY INSTRUMENT REVIEW FOR LWR PRESSURIZER VESSEL

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| PRESSURIZER (LIMITED TO INTERNALS & SPECIALITY ITEMS) | HEATER(S) PROBLEMS * THERMAL INDUCED: MECHANICAL WEAR, ELEMENT BURNOUT(S) | CFRs | | | | | | |
| (1) | | (General statement). | None. | None. | ? | Regulations are needed that address aging of pressure boundary components. | Regulations are needed that address license renewal. | The Code of Federal Regulations (CFRs) does not explicitly address the internals of the pressurizer or the vessel and supporting spray/surge piping. Since the pressurizer is part of the RCS pressure boundary, the rules set forth by the CFR principally apply to the vessel shell, the supports and spray/surge lines for design, fabrication, operation and preoperational testing. Applicable section noted for the pressurizer degradation issues are implied aging features for the internals and associated piping. |
| (2) | | 10 CFR 50, Appendix A, Criterion 14 | The RCPB shall be designed, fabricated and tested so a "low probability" of abnormal leakage and gross rupture exists. The criterion implies that the design shall account for mechanical degregation of the heater sheaths and sleeves. | None. | ? | Pressure boundary internals of the vessels should be addressed in the regulations. | Pressure boundary internals of the vessels should be addressed in the regulations. | Failure of the heater sheath and heater sleeve have resulted in pressure boundary leaks. (Heater burn-out is not a safety issue as these may be replaced. Heater leakage, however, is a safety issue.) |
| (3) | | TSs 3/4.4.3 | Provides mandatory restoration requirements for heater elements. | Mandatory surveillance could provide record for pressure cycles. | ? | N/A | Provides a record of heater replacement, i.e., operating history good or bad for the specific pressurizer. | Provides aging management by requiring replacement of heaters to maintain a level of power (heater) input capacity, i.e., the heaters must always perform at a prescribed level, thus forcing surveillance and repair. |
| (4) | | SRPs (General statement) | None. | None. | ? | SAR need to address failure mechanisms of the heaters, i.e., is replacement the only effective aging management procedure for the heaters? | Redundancy and ease of heater replacement should be evaluated for license renewal criteria. Testing of heaters should include excessive current leakage. | The pressurizer heaters are not explicitly addressed in the SRPs. Heater sleeve failures have resulted in pressure boundary RCS failures. Defective heaters and poor design are the probable causes for the pressure boundary failures; however, aging may be a factor in the failures, e.g., wearing and thinning due to rubbing action with supports caused by thermal growth. Although heaters are easy to replace, a technical safety issue exists because of the potential leakage path that could result in a primary containment leak. |
| (5) | | 5.2.4 | Requires ISI program to assess leaktight integrity. | None. | ? | Provides for selected (ASME Section XI) code inspection of welds for the life of the plant. | Verify that the inspection is adequate for license renewal. | |

General Notes:
1. A "?" indicates further study/investigation is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessarily the aging issue.
3. For meaning of abbreviations, acronyms, and initialisms, used throughout, see acronyms on page xi, xii, and xiii of the report.

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (6) | HEATERS (contd) | RGs (General statement) | None explicit. | None explicit. | ? | Revisions needed that explicitly address the aging of plant components and systems. | Revisions needed that explicitly address life extension and license renewal. | The NRC Regulatory Guides (RGs) do not explicitly address the internals of the pressurizer. |
| (7) | | ASME Code Section III (General) | None explicit. | None explicit. | ? | | Assurance that the current replacement methods/repair of failed heaters are adequate beyond the plant's license renewal date. | ASME codes requirements are primarily concerned with pressure boundary integrity; some sections of the code, however, cover other concerns, such as Section III NG that covers reactor core internals, not the pressure boundary. |
| (8) | | GISs GSI 13 | Extended overheating of heaters, cyclic fatigue, and service wear. | None. | None; issue was dropped. | Effects of overheating on pressurizer reliability. | Effects of pressurizer reliability on license extension. | Small break LOCA from extended overheating of pressurizer heaters. |
| (9) | | ANS 51.1-1983 PWR | Establishes the nuclear safety criteria and functional design requirements of structures, systems, and components of stationary NPPs. | Original design criteria could support LR. | ? | None. | None. | Operations, maintenance, and testing requirements are covered only to the extent that they affect design provisions. |
| (10) | | 58.11 | Provides design criteria for systems and equipment necessary to achieve and maintain a safe shutdown of the reactor to cold shutdown conditions from a hot standby or post accident condition. | Addresses safety functions that are closely aligned with aging if not correctly controlled, e.g., reactivity, RCS, heat removal, and RCS integrity including pressure and inventory control. | ? | None. | None. | |
| (11) | | IEEE 323 | Ability to perform safety function due to the effect of aging must be addressed. Types of aging include vibration and wear. | Acceptance and qualifying records. | ? | None. | None. | IEEE standards 323 as opposed to other instruments reviewed recognizes the need for aging and defines steps to address aging such as age conditioning and natural aging criteria. |
| (12) | | 741 | Provides criteria for protections requirements for Class IE power systems. Protection from electrical or mechanical damage or failures within a time frame too short for operator action. Doesn't include physical design factors such as fires, pipe whip, etc. | | | | | The protection refers to sense command and execute features, e.g., switch gear protection standby power protection and surge protection. Protection shall be designed to allow for periodic testing. |

-- Pressurizer Vessel

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | HEATERS (contd) | IEEE (contd) | | | | | | |
| (13) | | 934 | Provides criteria for replacement parts, both construction and operation. Requires inspection and testing of parts prior to release for service. | Acceptance testing and records that may be applicable to LR, e.g., failure review for wear, fatigue, original defects, insulation breakdowns, etc. | ? | None. | None. | Standard determines selection of required parts for Class 1E equipment, e.g., identification and failure review. However, required design changes are outside the scope of this standard. |
| (14) | | 344 | Establishes recommended practices to obtain data to qualify that the equipment will perform one SSE after an OBE. Included are tests for vibrational aging, seismic aging plus normal operation loads. | Acceptance and qualifying records. | ? | None. | None. | |
| (15) | SURGE & SPRAY LINES, NOZZLE PROBLEMS<br>* STRATIFIED FLOW, TRANSIENT THERMAL LOADS, THERMAL SHOCK; RESULTING IN LOW-CYCLE THERMAL FATIGUE | CFRs<br>10 CFR 50, Criterion 15 | RCS shall be designed for sufficient margin to assure that the design condition not be exceeded during normal operation or anticipated operational occurrences. This criterion implies that thermal shock and other factors in the design of the pressure boundary should be included. | None. | ? | Revision needed to aging or life of plant considerations. | Revision needed to address operations beyond the plant's license renewal date. | |
| (16) | | TSs<br>3/4.4.9.2 | Provides for control over the spray water ΔTs; restricts ΔT to a max. differential of 320°F. | Records for cycles. | ? | N/A | Could provide a record of component service applicable to license renewal, i.e., has the service exceeded TS temperature limits? | Provides aging management by restricting the temperature ΔTs. |
| (17) | | 3.9.3 | Addresses structural integrity of pressure retaining components. | Design in accordance with ASME III & 10 CFR-50, General Design Criteria 1,2,4,14 and especially GDC #15 that stipulates the design shall have sufficient margins such that operations including loading combinations and transients will not exceed the original design conditions. | ? | None. | Are the original design assumptions valid for license renewal? | |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (18) | SURGE & SPRAY LINES (contd) | TSs (contd) 3.9.1 | Addresses fatigue and stress and meeting the requirements of 10 CFR-50 GDC #15. Specifically address the acceptability of the design relative the number of cycles and events expected over the life of the plant. | Provides acceptance criteria applicable to license renewal. | ? | Verification of the number of cycles and events. | Adequate event records and operating history needed. Methods needed to predict the available fatigue life of the spray/surge lines at the end of the plant's license renewal date. | |
| (19) | | RGs 1.147 | No explicit reference to aging; however, provides an NRC acceptable list of approved ASME Code Cases for Section XI. | The RG implies life extension by providing acceptance method for approved (ASME) inspection methods using current Code Cases. | RG revisions are made to include or exclude appropriate Code Cases. | N/A | N/A | ASME Code Cases are not mandatory. This RG presents a NRC staff acceptable list of ASME council approved Code Cases for use in inspection and repair of components as required by 10 CFR 50, Appendix A, and Section 50.55a. |
| (20) | | 1.44 | Repair of RCPB lines. Process controls to minimize sensitization in stainless steel welds. | None. | None. | Better understanding of the relationship between welding process and sensitization is needed. | Determination of life expectancy of sensitized material and nuclear grade stainless steel is needed. | NRC program on "Evaluation of Welded and Repair-Weld Stainless Steel for LWR Service" initiated work in this area but was not completed. A small effort may be ongoing. |
| (21) | | ASME Code Section III | Addresses fatigue. | Fatigue curve revisions. | Section XI, SWG on Operating Plant Criteria is reviewing fatigue curves in Section III to determine if they can be revised to accommodate operation beyond 40 years. ASME SWG on PLEX has recommended that appropriate XI committees investigate more frequent NDE on surge line welds than currently required by Sec. XI, ISI. | Measurement of material fatigue life. | Improved end-of-life projections, 40 years and beyond. Accurate recording and records for pressure and thermal transients are needed to determine fatigue damage to spray and surge lines. | Studies are under way to assess the magnitude of the effects of actual environmental conditions. Code curves are based on smooth specimens in air at room temperature, whereas as materials in service have much rougher surfaces and are exposed to flowing coolant at operating temperatures. The actual surfaces are more prone to crack initiation. Studies and research to assess magnitude of effect of environmental factors could result in need for new Regulatory Guide, Regulatory instruments and/or code revisions. |
| (22) | * erosion (wall thinning) | NB-3210 Special considerations of NB-3121 (Section III) | Material subject to erosion, corrosion, and mechanical abrasion must have additional wall thickness to account for these anticipated conconditions. | None explicit; however design should provide guidance for the adequacy of the design beyond the 40-year interval. | ? | Implied in the design criteria. | Guidance is needed to cover the limits of component deterioration. | NB-3121 states "Material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provision made for these effects during the design or specified life of the component by a suitable increase in or addition to the thickness of the base metal over that determined by the design formulas."<br><br>Rules cover design and construction requirements but do not cover deterioration. |

-- Pressurizer Vessel

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (23) | SURGE & SPRAY LINES (contd) | GSI 47 | Thermal stress resulting from sustained operation of HPI. | None. | None. | ? | Effects on pressurizer reliability at plants at which pressurizer has been over heated and cooled. | NRC concluded there is no direct safety implication of overfeeding and over-cooling pressurizer with HPI. |
| (24) | | ANS 51.1-1983 PWR | Establishes the nuclear safety criteria and functional design requirements of structures, systems, and components of stationary NPPs. | Original design criteria could support LR. | ? | None. | None. | 51.1 is for PWRs and 52.1 is for BWRs. The standards are essentially the same. Operations, maintenance, and testing requirements are covered only to the extent that they affect design provisions. |
| (25) | | 58.11 | Provides design criteria for systems and equipment necessary to achieve and maintain a safe shutdown of the reactor to cold shutdown conditions from a hot standby or post accident condition. | Addresses safety functions that are closely aligned with aging if not correctly controlled, e.g., reactivity, RCS heat removal, and RCS integrity including pressure and inventory control. | ? | None. | None. | |
| (26) | SPRAY HEAD<br>* THERMAL INDUCED, EMBRITTLEMENT<br>* EROSION | CFRs<br>10 CFR 55a, Codes and Standards | Regulation refers to ASME Section III for design of class 1 components. Implied that erosion will be addressed in piping system design. | None. | ? | Revision needed to address aging. | Revision needed to address operation beyond the plant's license renewal date. | |
| (27) | | TSs<br>3/4.4.9.2 | Provides for specific ΔT (100°F) heatup and ΔT (200°F) cooldown in any 1-hour period and max spray water differential at 320°F. | Records for cycles. | ? | N/A | Could provide a record of component service applicable to license renewal, i.e., has the service exceeded TS temperature limits? | Provides aging management by restricting the temperature ΔTs. |
| (28) | | Section III | None. | None. | ? | None. | None. | ASME code requirements are exclusively concerned with pressure boundary integrity. |
| (29) | | NB-3210 Special considerations of NB-3121 (Section III) | Material subject to erosion, corrosion, and mechanical abrasion must have additional wall thickness to account for these anticipated conditions. | None explicit; however, design should provide guidance for the adequacy of the design beyond the 40-year interval. | ? | Implied in the design criteria. | Guidance is needed to cover the limits of component deterioration. | NB-3121 states "Material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provision made for these effects during the design or specified life of the component by a suitable increase in or addition to the thickness of the base metal over that determined by the design formulas."<br><br>Rules cover design and construction requirements but do not cover deterioration. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (30) | SPRAY HEAD (contd) | GSI 47 | Thermal stress resulting from sustained operation of HPI. | None. | None. | ? | Effects on pressurizer reliability at plants at which pressurizer has been overfed and cooled. | NRC concluded there is no direct safety implication of overfeeding and over-cooling pressurizer with HPI. |
| (31) | | ANS<br>51.1-1983 PWR<br>52.1-1983 BWR | Establishes the nuclear safety criteria and functional design requirements of structures, systems, and components of stationary NPPs. | Original design criteria could support LR. | ? | None. | None. | 51.1 is for PWRs and 52.1 is for BWRs. The standards are essentially the same. Operations, maintenance, and testing requirements are covered only to the extent that they affect design provisions. |
| (32) | | 58.11 | Provides design criteria for systems and equipment necessary to achieve and maintain a safe shutdown of the reactor to cold shutdown conditions from a hot standby or post accident condition. | Addresses safety functions that are closely aligned with aging if not correctly controlled, e.g., reactivity, RCS heat removal, and RCS integrity including pressure and inventory control. | | | | |
| (33) | SHELL INTERNAL<br>* SHELL BARREL IN STEAM SPACE: HIGH & LOW CYCLE THERMAL BENDING AT WATER TO STEAM INTERFACE | CFRs<br>10 CFR 55a, Codes and Standards | Regulation references design in accordance with ASME Code Section III for class 1 components. Implied that the design will address high- and low-cycle fatigue, bending stress, etc. | None. | ? | Regulation revision needed to address aging. | Regulation revision needed to address operations beyond the plant's license renewal date. | |
| (34) | | TSs<br>3/4.4.9.2 | Provides for specific $\Delta T$ (100°F) heatup and $\Delta T$ (200°F) cooldown in any 1 hour period and max. spray water differential at 320°F. | Records for cycles. | ? | N/A | Could provide a record of component service applicable to license renewal, i.e., has the service exceeded TS temperature limits? | Provides aging management by restricting the temperature $\Delta T$s. |
| (35) | | 2.1.1 | Establishes mandatory high-temperature limit for the pressurizer (highest reactor coolant temperature). | Auto trip provides records of transient conditions. | ? | N/A | Could provide a record of transient conditions and monitoring of cyclic events. | Provides aging management by establishing the upper material temperature limits. |
| (36) | | 2.1.2 | Establishes mandatory high-pressure limit for the reactor coolant system. | Auto trip provides records of transient conditions. | ? | N/A | Could provide a record of transient conditions and monitoring of cyclic events. | Provides aging management by establishing the high-pressure limits. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (37) | SHELL INTERNAL (contd) | SRPs 3.9.3 | Addresses structural integrity of pressure retaining components. | Design in accordance with ASME III and 10 CFR 50 General Design Criteria 1, 2, 4, 14, 15, and especially GDC #15 that stipulates the design shall have sufficient margins such that that operations including loading loading combinations and transient will not exceed the original design conditions. | ? | None. | Are the original design assumptions valid for license renewal? | |
| (38) | | 5.2.2 | Overpressure protection. | Overpressure protection. | ? | N/A | Probably adequate for license renewal for this specific event. Requires protection for the "life of plant" events. | Review stipulates that overpressure protection must be available (safety/relief valves) for the pressurizer @110% of allowable pressure for transients or operational occurrences one or more times during the life of the plant. |
| (39) | | 3.9.1 | Addresses fatigue and stress and meeting the requirements of 10 CFR 50 GDC #15. Specifically address the acceptability of the design relative the number of cycles and events expected over the life of the plant. | Provides acceptance criteria applicable to license renewal. | ? | Verification of the number of cycles and events. | Adequate event records and operating history needed. Methods are needed to predict the available fatigue life of the pressurizer shell at the end of the plant's license renewal date. | |
| (40) | | ASME Code Section III | Requalification of components. | Requalification of components. | A new appendix is under consideration which may be used to requalify components which have exceeded the rules for cyclic operation in Section III, NB-3222.4. A task group has been formed to address the issue. This also applies to Section XI. | Evaluate new appendix. | Requalification Rules for components. | |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (41) | SHELL INTERNAL (contd) | ASME Code (contd) | Fatigue. | Fatigue curve revisions. | Section XI, SWG on Operating Plant Criteria is reviewing fatigue curves in Section III to determine if they can be revised to accommodate operation beyond 40 years. | None. | Improved end-of-life projections, 40 years and beyond. | Studies are under way to assess the magnitude of the effects of actual environmental conditions. Code curves are based on smooth specimens in air at room temperature, whereas materials in service have much rougher surfaces and are exposed to flowing coolant at operating temperatures. The actual surfaces are more prone to crack initiation. Studies and research to assess magnitude of effect of of environmental factors could result in need for new Regulatory Guide, Regulatory instruments and/ or code revisions. |
| (42) | | Section XI | Condition assessment. | Requalification of systems and components. | Consideration of development of new Article IWX-8000, Requalification for Fatigue life. An ASME WG has been formed on Fatigue in NPPs. The goal of the WG is to provide a mechanism for relief when the fatigue design limit is reached. (The limit would be determined by developing monitoring and evaluation techniques.) | Condition assessment, Information/data. | Requalification of systems and components following expiration of operating license. | |
| (43) | | GSI 47 | Thermal stress resulting from sustained operation of HPI | None. | None. | ? | Effects on pressurizer reliability at plants at which pressurizer has been overfed and cooled. | NRC concluded there is no direct safety implication of overfeeding and over-cooling pressurizer with HPI |
| (44) | | ANS 51.1-1983 PWR 52.1-1983 BWR | Establishes the nuclear safety criteria and functional design requirements of structures, systems, and components of stationary NPPs. | Original design criteria could support LR. | ? | None. | None. | 51.1 is for PWRs and 52.1 is for BWRs. The standards are essentially the same. Operations, maintenance, and testing requirements are covered only to the extent that they affect design provisions. |
| (45) | | 58.11 | Provides design criteria for systems and equipment necessary to achieve and maintain a safe shutdown of the reactor to cold shutdown conditions from a hot standby or post accident condition. | Addresses safety functions that are closely aligned with aging if not correctly controlled, e.g., reactivity, RCS heat removal, and RCS integrity including pressure and inventory control. | ? | None. | None. | |

-- Pressurizer Vessel

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (46) | THERMAL SLEEVE(S) * THERMAL STRESS: FATIGUE | TSs 3/4.4.9.2 | Provides for specific ΔT (100°F) heatup and ΔT (200°F) cooldown in any 1-hour period and max. spray water differential at 320°F. | Records for cycles. | ? | N/A | Could provide a record of component service applicable to license renewal, i.e., has the service exceeded TS temperature limits? | Provides aging management by restricting the temperature ΔTs. |
| (47) | | 2.1.1 | Establishes mandatory high-temperature limit for the pressurizer (highest reactor coolant temperature). | Auto trip provides records of transients conditions. | ? | N/A | Could provide a record of transient conditions and monitoring of cyclic events. | Provides aging management by establishing the upper material temperature limits. |
| (48) | | 3.9.1 | Addresses fatigue and stress and meeting the requirements of 10 CFR 50 GDC #15. Specifically address the acceptability of the design relative the number of cycles and events expected over the life of the plant. | Provides acceptance criteria applicable to license renewal. | ? | Verification of the number of cycles and events. | Adequate event records and operating history needed. Methods are needed to predict the available fatigue life of the pressurizer shell at the end of 40 years. | |
| (49) | | Section XI | Fatigue. | Fatigue curve revisions. | Section XI, SWG on Operating Plant Criteria is reviewing fatigue curves in Section III to determine if they can be revised to accommodate operation beyond 40 years. | Measurement of material fatigue life. | Improved end-of-life projections, 40 years and beyond. Accurate recording and records for pressure & thermal transients are needed to determine fatigue damage to spray and surge lines. | Studies are under way to assess the magnitude of the effects of actual environmental conditions. Code curves are based on smooth specimens in air at room temperature, whereas materials in service have much rougher surfaces and are exposed to flowing coolant at operating temperatures. The actual surfaces are more prone to crack initiation. Studies and research to assess magnitude of effect of environmental factors could result in need for new Regulatory Guide, Regulatory Instruments and/or code revisions. |
| (50) | | ANS 51.1-1983 PWR 52.1-1983 BWR | Establishes the nuclear safety criteria and functional design requirements of structures, systems, and components of stationary NPPs. | Original design criteria could support LR. | ? | None. | None. | 51.1 is for PWRs and 52.1 is for BWRs. The standards are essentially the same. Operations, maintenance, and testing requirements are covered only to the extent that they affect design provisions. |
| (51) | CORROSION * SHEATHES: CHEMICALLY INDUCED IGSCC & FATIGUE. * GENERAL: BORATED COOLANT LEAKS * BOLTING/CLOSURES: SCC, LUBRICATION, MOISTURE ENVIRONMENT | CFRs 10 CFR 55a, Codes and Standards | Regulation requires design, fabrication and construction in accordance ASME Code Section III, class 1 components. Code requires attention given to corrosion, erosion, environmental effects; this implies the regulation addresses the aging issue. | None. | ? | None. | None. | Improvements in the ASME Code are needed to address deterioration of materials. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (52) | CORROSION (contd) | SRPs<br>5.2.3 | Materials are reviewed for suitability for the service. Review of manufacturing and welding is also included. | None. | ? | None. | None. | |
| (53) | | RGs<br>1.45 | Defines leak detection methods for RCPB leaks that monitor potential changes in containment environment, such as increasing humidity. | Should provide records of past plant conditions and current leak status for evaluation of license removal applications. | ANL has developed an acoustic method for leak monitoring. | Is the current RG position adequate for aging management? | Is the current RG position adequate for license renewal? | The ANL acoustic laboratory experimental methods need to be verified by field tests. |
| (54) | | ASME Code<br>NB-3210 Special considerations of NB-3121 (Section III) | Requirements implied for the specific pressurizer parts. The intent of the code is base material subject to erosion, corrosion, and mechanical abrasion that must have additional wall thickness to account for the anticipated conditions. | None explicit; however, design should provide guidance for the adequacy of the design beyond the 40-year interval. | ? | Implied in the design criteria. | Guidance is needed to cover the limits of component deterioration. Adequate monitoring is needed to detect boric acid leakage and corrosion. | NB-3121 states "Material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provision made for these effects during the design or specified life of the component by a suitable increase in or addition to the thickness of the base metal over that determined by the design formulas."<br><br>Rules cover design and construction requirements but do not cover deterioration. |
| (55) | | GSI A-16 | Plateout of activation products; increase in occupational doses. | None. | Issue was resolved upon issuance of guidance to utilities. | Long-term effects of chemical decontamination agents on the primary coolant system. | | See NUREG/CR-2963. |
| (56) | | ANS<br>51.1-1983 PWR | Establishes the nuclear safety criteria and functional design requirements of structures, systems, and components of stationary NPPs. | Original design criteria could support LR. | ? | None. | None. | Operations, maintenance, and testing requirements are covered only to the extent that they affect design provisions. |
| (57) | | 58.11 | Provides design criteria for systems and equipment necessary to achieve and maintain a safe shutdown of the reactor to cold shutdown conditions from a hot standby or post accident condition. | Addresses safety functions that are closely aligned with aging if not correctly controlled, e.g., reactivity, RCS heat removal, and RCS integrity including pressure and inventory control. | ? | None. | None. | |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (58) | SURVEILLANCE:<br>* TESTING<br>* INSPECTION<br>* MAINTENANCE & REPAIRS<br>* NDE<br>* INFORMATION/DATA<br>-FREQUENCY<br>-DATA COLLECTION<br>-DATA EVALUATION<br>-TREND CURVES<br>-RECORDING KEEPING | CFRs<br>10 CFR-50, Appendix A, Criterion 32 | Regulation requires the RPBC be designed to facilitate periodic inspections. | Inspection records could establish integrity or condition of the component. | ? | N/A | Verify that inspection records are adequate for license renewal. | |
| (59) | | 10 CFR-50, Appendix B | Regulation requires a QA plan for design, operations and testing. | QA documentation will assist in license renewal process. | ? | N/A | Verify that the QA documentation is adequate for current license renewal needs. | |
| (60) | | 10 CFR-55a, Codes and Standards | Regulation requires ISI in accordance with ASME Section XI. | ISI records will assist in license renewal processes. | ? | N/A | ISI intervals and exception allowed may not fully address license renewal concerns or requirements. | |
| (61) | | TSs<br>3/4 4.0.2 & 4.0.5 | Provides surveillance intervals and references ASME Code Section XI for class 1 components. | Component status records. | ? | N/A | Records could be used for license renewal, i.e., verify the history of component integrity. | Should provide history of component for license renewal application. |
| (62) | | 3/4.4.3 & including 4.4.3.1 4.4.3.2 4.4.3.3 | TS with its surveillance requirements dictate the electrical heater capacity and water level of the pressurizer. | Component status records. | ? | N/A | Records could be used for license renewal, i.e., verify the history of component integrity. | |
| (63) | | 6.0 (Administrative controls) 6.10.1 | Lists mandatory 5-year retention records list. | Provides record history of components and systems. | | N/A | May require longer retention period than 5 years for principal inspections, item 6.10.1.b. and d. to be applicable for license renewal. | |
| (64) | | 6.10.2 | Lists mandatory life of plant period for plant retention records. Principal interest is item 6.10.2 e. which specifies the record keeping time for transients or operating cycles. | Provides record history of components and systems. | ? | N/A | Could provide sufficient information and records for QA, ISI, water quality and others, including specific records for component transient conditions, i.e., TS table 5.7.1 "Component Cyclic or Transient Limits." | |
| (65) | | SRPs<br>5.2.4 | Requires ISI program to assess leaktight integrity. | None. | ? | Provides for selected (ASME Section XI) code inspection of welds for the life of the plant. | Is inspection adequate for license renewal? | Inspection are to be in accordance with 10 CFR-50, Appendix A, Criterion 32 and 10 CFR-55a and as detailed in ASME Section XI. |
| (66) | | 17.2 | Addresses maintenance and testing relative the implementation of a Quality Assurance plan. | Records of maintenance and testing. | ? | Is in-place QA plan and record keeping adequate for aging management records? | Is the QA plan adequate for license renewal? | The SRP does not explicitly address aging or license renewal, only that a QA plan is required. |

-- Pressurizer Vessel

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (67) | SURVEILLANCE (contd) | RGs 1.58 | Establishes guidance for the qualification of NDE staff who perform inspections, tests and examinations for NPPs. | Provides documentation of staff qualifications and quality assurance records (of staff) for the plant QA plan, i.e., records needed for license renewal. | ? | Improved qualification guidance may be necessary to fully address aging concerns. | Improved qualification guidance may be necessary for LR. | References ANSI N45.2.6-1978. |
| (68) | | Safety Guide 30 (from RG 1.30) | Establishes regulatory position for QA requirements relative to testing of electrical equipment. | QA documentation. | ? | None. | None. | References 10 CFR 50, Appendix B, QA criteria for NPPs. |
| (69) | | ASME Code Section XI (general) | Material and component condition assessment. | Ref. info./data bases as they apply to requests for extensions/renewals. | New nonmandatory appendix on recordkeeping has passed the SWG-PLEX. | See other columns. | Ref. Info/Data Bases to support extensions/renewals. | As materials and components age, the predictive capabilities for physical and mechanical property changes must improve. Since, during a plant's service life, an increasing amount of information/data will be available from surveillance programs, better procedures to evaluate and use this information/data can and must be developed. Stratified flow and thermal shock are major stressors in the base metal and welds. Weld metal inspections alone are insufficient for mechanical property changes. |
| (70) | | Section XI | Surveillance. | New baseline examination requirements have been established. | A new task group on baseline examination. | See other columns. | Possible need to require a new baseline examination for a license extension request. | |
| (71) | | | Allows for inspections beyond 40 years. | Changes to Inspection Plans A and B which are currently based on a 40-year operating life based on four intervals, the sum of which = 40. | SWG has implemented revisions to IWA-2400, which will delete the 40-year limit currently contained in Section XI. Also, reviewing need for more frequent and extensive inspections. inspections. ASME PLEX SWG is also reviewing inspection Plan A for application to PLEX or should a new plan (of higher reliability) be developed. | Extended and/or enhanced surveillance beyond 40 years. | Surveillance beyond 40 years. | The ASME PLEX SWG is reviewing the Inspection Plan A to determine if it should be used. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| (72) | SURVEILLANCE (contd) | ASME Code (contd) | Monitoring and testing. | Information/Data for trend curves. | Development of new exam. techniques to detect age-related degradation and fatigue being considered. | Information, data, and assessments. | Monitoring/testing techniques. | |
| (73) | | | Information and data for trend curves projections. | Records, Informa-Data. | SWG is considering development of non-mandatory appendix for recordkeeping. | Information and data for trend curves. | Records, information, and data. | This Appendix would provide utilities with guidance as to the records needed to support a license extension request. |
| (74) | | | Fatigue failures. | Anticipate and avoid fatigue failures. | Fatigue monitoring technology developed and demonstrated; several utilities proceeding. | Experience and test data for high-cycle effects. | Regulatory acceptance of alternate approach to design bases. Addition of fatigue curves to high cycles is needed for life extension. | a) Failures unlikely from cycles considered in design. Design approach is conservative. Component features included to preclude fatigue failure. Components have high tolerance to flaws.<br>b) Fatigue cracking has occurred during operating:<br>• vibration<br>• rapid thermal cycling<br>• pre-existing flaws<br>• other conditions not considered in design. |
| (75) | | Table IWB-2500 | Weld integrity of heater penetration welds require visual inspections. | Records and acceptance standards. | None. | Frequency of examination. | Justify the change in ISI interval. (The existing frequency may not be sufficient for license renewal.) | Section XI in Table IWB-2500 specifically addresses the pressurizer; however, only the pressure vessel, piping, nozzles, bolts and hydrostatic/leakage tests are addressed. The code does not address the pressurizer internals. |
| (76) | | GISs<br>GSI B-47 | Long-term degradation of Class 1, 2, 3 component supports. | ISI requirements. | Issue was dropped from further consideration. | Characterize long-term degradation of component supports. | Level of ISI inspection needed to verify adequacy of supports. | NRC referenced ASME Code, Section XI and QA Program before dropping issue. |
| (77) | | ANS<br>3.1 | Provides criteria for selection, qualification, and training of personnel for stationary nuclear power plants. | Provides record on training of staff. | ? | None. | None. | |
| (78) | | 3.2 | Provides recommendations and requirements for administrative controls, including written procedures, and QA program to help assure that activities of NPPs are carried out without risk to health and safety of the public. | Administrative in-place controls will probably be required for LR. The associated QA documentation will be useful for LR. | ? | None. | None. | Among the activities covered under this standard are design changes, fabricating, cleaning-decontamination, inspecting, testing, maintaining, and repairing. |

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | ANS (contd) | | | | | | |
| (79) | | 51.1-1983 PWR | Establishes the nuclear safety criteria and functional design requirements of structures, systems, and components of stationary NPPs. | Original design criteria could support LR. | ? | None. | None. | Operations, maintenance, and testing requirements are covered only to the extent that they affect design provisions. |
| (80) | | 58.11 | Provides design criteria for systems and equipment necessary to achieve and maintain a safe shutdown of the reactor to cold shutdown conditions from a hot standby or post accident condition. | Addresses safety functions that are closely aligned with aging if not correctly controlled, e.g., reactivity, RSC heat removal, and RCS integrity including pressure and inventory control. | ? | ? | Is the design criteria valid for LR? | |
| | | IEEE | | | | | | |
| (81) | | 336 | Provides minimum requirements for inspection and testing of Class IE power, instrumentation and control equipment during construction phase. | Records of construction may be applicable to LR. | ? | None. | None. | Provides a criteria for correct installation which could effect aging aspects of the equipment after start-up and operation. Class IE is equipment that is essential to emergency shutdown, containment isolation, RCC, and CHR. |

# APPENDIX VI

## REGULATORY INSTRUMENT REVIEW FOR EMERGENCY DIESEL GENERATOR

# Understanding and managing aging of emergencey diesel generators

**Principal Diesel Engines in Nuclear Service**

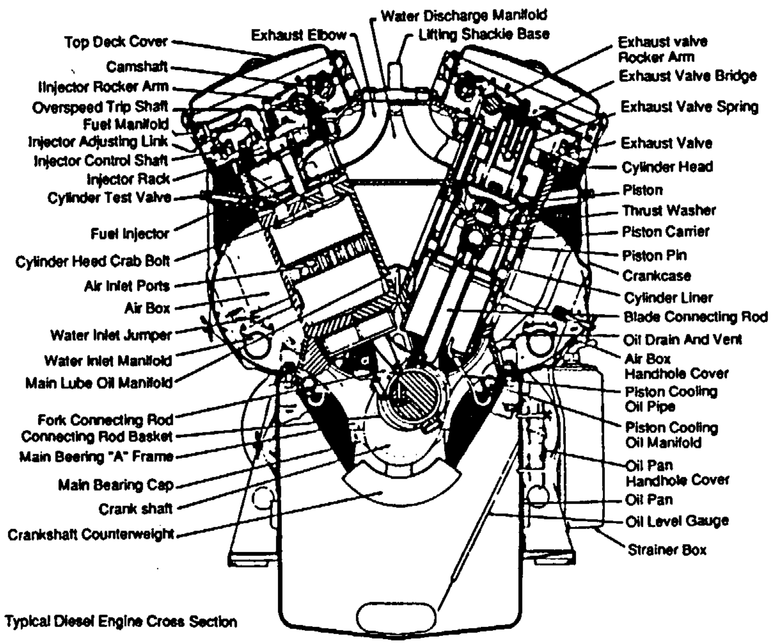
Manufacturer

ALCO
Allis Chalmers
Caterpillar
Cooper Bessemer
Fairbanks Morse
Electro-Motive Division
Nordberg
Transamerica Delaval
Worthington
Others

Materials: (typical)
Alloy steels, welded steel plates, castiron includng, gray, aluminum, stellite seats, forged steels, ductive irons, non metallic gaskets, hoses, seals

Capacities:
HP 215 to 670 per cylinder or 800 HP to 8390 HP
KW ratings 50, 500, 1200, 3000, 6000

Stressors:
Cooling water, lubricating oil, fuel oil, starting air, intake and exhaust, deterioration, dynamic stress, vibration, thermal fatigue, wear and harsh testing




Typical Diesel Engine Cross Section

| UNDERSTANDING AGING (Stressor & Enviroment Interations) | | MANAGING AGING | |
|---|---|---|---|
| Sites | Aging Concerns | Inservive Inspection, Serveillance | Mitigation |
| • Instrument and control systems<br>Governor<br>Sensors<br>Relays<br>• Startup component<br>• Fuel system<br>Piping on engine<br>Injector pumps<br>• Starting system<br>Controls<br>Starting air valve<br>Starting motors<br>Air compressor<br>• Switchgear system<br>Breakers<br>Relays<br>Instrument and controls<br>• Cooling system<br>Pumps<br>Heat exchangers<br>Piping<br>• Lubricating system<br>Heat exchangers<br>Pumps<br>Lube oil | • Environmentally induced: dust, water, heat, oil, chemical, etc.<br>• Maintenance errors: inadequate training, maladjustments, etc.<br>• Fast starts and other regulatory induced factors<br>• Design inadequacy, wrong application, or poor component<br>• Operation induced: inadequate training and skills<br>• Vibration induced<br>• Fuel or lubrication degeneration<br>• Gasket, seal, or organic material degeneration<br>• Inadequate spares: quality, strorage, ordering problems, data and specifications<br>• Corrosion, oxidation<br>• Thermal stress<br>• Manufacturing or quality problems<br>• Fatigue not related to vibration<br>• Metal fatigue<br>• Wear<br>• Bacterial action | NRC Requirements<br>• RC 1.9...surveillance, maintenance, periodic testing<br>• RG 1.108...routine testing, maybe with drawn due to RG 1.9 revisions<br>• General letter 83.41..."fast cold" starts<br>• 10-CFR-50, appendix A, criterion 4, 5, 17, 18 & 50 ... periodic testing<br>• 10-CFR-5, appendix B, section XI requires... components & system to perform satisfactorily<br>• 10-CFR-50, 55a Codes & St'd - ASME BPVC section III, IX, IEEE STD 279<br>• TS 3/4 8.1 surveillance/inspection for operation & shutdown status<br>• GSI B-56-improve reliability of Eng.<br>• IEEE standard acceptable for use by NRC-RG 1.9 | Integrated EDG program of testing, inspection, monitoring, trending and maintenance activities:<br>• Testing/trending, change testing to a slower start test and acquisition of these testing parameters for trending<br>• Improved inspection of weekly, monthly, and yearly to determine envioronmental stressors more effectively<br>• Maintenance responsive to test & inspection, e.g., do not over haul unless inspection and trends indicate the need<br>• Increased training for EDG Staff in on-site maintenance<br>• Systen modifications to mitigate stressors |

# REGULATORY INSTRUMENT REVIEW FOR EMERGENCY DIESEL GENERATORS

| COMPONENT COLUMN 1 | ISSUE COLUMN 2 | REG. INSTRUMENT COLUMN 3 | AGING FEATURES COLUMN 4 | LIFE EXTENSION FEATURES COLUMN 5 | CURRENT INITIATIVES COLUMN 6 | AGING NEEDS COLUMN 7 | LIFE EXTENSION NEEDS COLUMN 8 | COMMENTS COLUMN 9 |
|---|---|---|---|---|---|---|---|---|
| EMERGENCY DIESEL GENERATOR (1) | CYCLIC FATIGUE, INCLUDING VIBRATION AND SERVICE WEAR | CFRS 10 CFR 50 Appendix B | Section XI of Appendix B requires test program to assure that components and system perform correctly. | Provides records and performance history for EDGs. | None. | None. | Are test documents/ records adequate for LR? | Requires that written test procedures that address proof test prior to installation, preoperational tests and operational tests. |
| (2) | | Appendix A | General Design Criteria including criterion 2, 4, 5, 17, 18, and 50. Of these, 18 is most important as it provides aging guidance by requiring periodic testing and inspection to evaluate component condition. | Provides records and performance history for EDGs. | Interpretation of this CFR may change if the suggested revisions to 1.9 are approved. | None. | None. | 10 CFR 50, Appendix A GDC, criterion 17, requires an onsite EDG to provide onsite power to safety-related components and systems. |
| (3) | | SRPs 9.5.7 | EDG lub oil system - review addresses dry starts or lack of lub during starts. The SRP requires a dedicated system for wearing parts. | None. | None. | None. | The records (history) of the EDG should indicate adequate performance to permit use beyond 40 years. | |
| (4) | | RGs 1.32 | Regulatory position for design, construction and operation of NPPs; aging features implied. | None. | None. | None. | See comment. | Refers to IEEE standard 308-74 as acceptable criteria for design, general operation, and testing of NPPs. The RG does not address license renewal. |
| (5) | | ASME Code Section III Section XI | EDGs by definition are Class 1, 2, and 3 components; however, the EDGs are not pressure vessels or piping, etc. | None. | N/A | None. | None. | The ASME codes as required in 10 CFR 50.55a mandate ISI. These requirements rarely involve EDG unless a specific part is designed and manufactured in accordance with the Code. |
| (6) | | IEEE 308 | Design base criteria for Class IE equipment to enable them to meet their functional requirements. | Provides an acceptable design base for possible LR. | None. | None. | None. | Includes power systems, e.g., diesel generators: design base includes malfunctions, accidents or operating modes that could lead to degradation of the systems. Malfunctions are defined as natural phenomena, e.g., environmental factors of pressure, humidity, temperatures, and accidents, e.g., fires. |
| (7) | | 323 | Ability to perform safety function due to the effect of aging must be addressed. Types of aging include vibration and wear. | Acceptance and qualifying records. | None. | None. | None. | IEEE standard 323 as opposed to other instruments reviewed recognizes the need for aging and defines steps to address aging such as age conditioning and natural aging criteria. |

General Notes:
1. A "?" indicates further study/investigation is needed.
2. For the GSIs, "resolved" means the generic safety issue is resolved, not necessarily the aging issue.
3. For meaning of abbreviations, acronyms, and initialisms, used throughout, see acronyms on page xi, xii, and xiii of the report.

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|---|---|---|---|---|---|---|---|---|
| (8) | | IEEE (contd) 387 | Provides principal design criteria for EDGs to meet their functional requirements under design base conditions. | Provides an acceptable design base for possible LR. | IEEE 387 is currently being revised. Will incorporate IEEE Standard 749 requirements. IEEE-387 already mentions aging and aging management techniques. | None. | ? | Provides the minimum service requirements for the design staff (A&E) including operational cycles, operational hours, temperatures (max. and min.), seismic response, radiation, humidity, air quality etc., i.e., all the conditions that should be considered in the design. Fortyseven (47) design and application considerations are given in the standard. |
| (9) | EXTERNAL ENVIRONMENTAL FACTORS, INCLUDING:<br>• CHEMICALS<br>• HUMIDITY<br>• OILS AND COMBUSTION PRODUCTS<br>• DUST AND FOREIGN MATERIAL ON RELAYS/CONTACTS<br>• NATURAL PHENOMENA-EARTHQUAKES, FIRE AND FLOODS, ETC. | CFRs 10 CFR 50.49 | Provides that environmental specifications be submitted for electrical safety-related equipment that address environmental factors, e.g., aging, radiation, temperatures, and humidity, and other conditions, e.g., loss of ventilation, vibrations, and pipe breaks. | Addresses aging and degradation issues applicable to conditions of plant for LR. | ? | None. | Are the specification sufficient for current real-time conditions? | |
| (10) | | 10 CFR 50, Appendix A | GDC of Appendix A including 2 and 4 require design base for natural phenomena, e.g., floods and earthquake, and missiles, e.g., EDG must be compatible with normal operation, accidents and postulated missiles, and pipe whip. | Address natural phenomena. | N/A | None. | Is the plant's current status the same or has a change in seismic category developed since original license? | |
| (11) | | SRPs 2.4.2 | Refers to 10 CFR 50 GDC, Criterion 2, "components important to safety being designed to withstand the effects of hurricanes, floods, tsunami, and seiches." Also refers to 10 CFR 100 for identifying and evaluating hydrologic features of the site. | Original design considers flood design and local precipitation, i.e., this will provide records history. A flood history up-date is also required. | None. | None. | Was the original design adequate for current conditions, i.e., has the history changed? | The CFR principally provides the criteria for the original design. The CFR does not address the aging issues. |

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|---|---|---|---|---|---|---|---|---|
| (12) | EXTERNAL ENVIRONMENTAL FACTORS (contd) | SRPs (contd) 3.2.1 | Refers to 10 CFR 50, Appendix A, GDC, Criterion 2, for seismic classification to withstand earthquakes without a loss of ability to perform safety functions. Also refers to 10 CFR 100, Appendix A, for the performance (SSE), including standby diesel generator auxiliary systems. | None. | None. | None. | Verify that the components seismic categories have not changed. | The CFR principally provides the criteria for the original design. The CFR does not address the aging issue. |
| (13) | | 3.5.1.1 3.5.1.2 | Provides for review and acceptance of internal and external missile protection for safety required for safe shutdown of the plant. This ultimately mitigates damage or degradation to the EDG system from missiles. | The protection must be for the life of the plant. | None. | None. | None, unless design base has changed to warrant additional protection from missiles. | These requirements are principally start-up requirements, but remain in effect for the life of the plant. |
| (14) | | 3.11 | Provides for acceptance criteria stipulating that the mechanical component will perform satisfactorily for the "length of time" for which its function is required for harsh and mild environments. | Provides records for environmental acceptance and qualification. | N/A | Should be specific in stating that acceptance should be maintained over the life of plant. | Acceptance criteria needed for LR. | This is principally an acceptance criteria standard; the SRP does not address the life of plant issues. |
| (15) | | 7.1 | Provides acceptance criteria of instrumentation and controls. Refers to 10 CFR 50, GDC and IEEE 279 for design basis for natural phenomena and environmental concerns. | Provides acceptance criteria records. See table 7.1, "Acceptance Criteria." | None. | Should be specific in stating that acceptance should be maintained over the life of plant. | Acceptance criteria needed for LR. | Although the SRP does not discuss aging or LR, the SRP requirements should be useful to aging management and LR if adequate records are maintained for the life of the plant. |
| (16) | | 9.5.4 | EDG (starting) engine fuel oil - review determines the quality of the engine fuel oil. System should be free from oil degradation to prevent engine failure. | Provides acceptance criteria and records. | None. | None. | Acceptance criteria needed for LR. | Although the SRP does not discuss aging or LR, the SRP requirements should be useful to aging management and LR if the requirements are met and adequate records are maintained for the life of the plant. |
| (17) | | 9.5.6 | EDG engine start-system - review requires determining the adequacy of the quality and condition of the air supply. Clean, dry air required. | Provides records for system status. | None. | None. | Acceptance criteria needed for LR. | Although the SRP does not discuss aging or LR, the SRP requirements should be useful to aging management and LR if the requirements are met and adequate records are maintained for the life of the plant. |

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|---|---|---|---|---|---|---|---|---|
| (18) | EXTERNAL ENVIRONMENT FACTORS (contd) | SRPs (contd) 9.5.7 | EDG engine lub system - lub oil temperature must be maintained. | System records. | | | | |
| (19) | | RGs 1.29 | EDG must be designed for seismic category 1 for SSE occurrences. | None. | None. | N/A | The original design must be adequate for current known seismic conditions or postulated conditions. | |
| (20) | | 1.100 | Seismic qualifications, principles, procedures and methods. | None. | None. | N/A | LR questions should be addressed in the RG. | Refers to IEEE Standard 344-87. |
| (21) | | ASME Code Section III Section XI | EDGs by definition are Class 1, 2, and 3 components; however, the EDGs are not pressure vessels or piping, etc. | None. | None. | None. | None. | The ASME codes as required in 10 CFR 50.55a mandate ISI. These requirements rarely involve EDG unless a specific part is designed and manufactured in accordance with the Code. |
| (22) | | ANS 2.2 | Minimum instrumentation for system input of ground motion to provide evaluation of data whether or not vibratory motions have been exceeded. | Instrumentation should provide records of plant service. | N/A | None. | None. | |
| (23) | | 2.8 | Methodology is described to evaluate the flood having virtually no risk of exceedance that may be caused by precipitation and snowmelt and dam failures. | None. | N/A | None. | None. | |
| (24) | | 2.10 | Defines the type of timing of plant owner activities required in the event of an earthquake and includes specific procedures for the evaluation of records obtained from seismic instrumentation specified in ANS-2.2-1978. | Records of plant activities that may assist in LR. | N/A | None. | None. | |
| (25) | | 2.12 | Provides guidelines that allow reactor designers to select hazards, e.g., natural, man-made or combinations of hazards, that effect the design of system and components. | None. | N/A | None. | None. | |

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|---|---|---|---|---|---|---|---|---|
| (26) | EXTERNAL ENVIRONMENTAL FACTORS (contd) | IEEE 308 | Design base criteria for Class 1E equipment to enable them to meet their functional requirements. | Provides an acceptable design base for possible LR. | ? | None. | None. | Includes power systems, e.g., diesel generators; design base includes malfunctions, accidents or operating modes that could lead to degradation of the systems. Malfunctions are defined as natural phenomena, e.g., earthquakes and floods, and postulated phenomena, e.g., environmental factors of pressure, humidity, temperatures, and accidents, e.g., fires. |
| (27) | | 323 | Addresses ability to perform safety function due to aging, including factors of natural and environmental conditions that must be addressed. | Acceptance and qualifying records. | ? | None. | None. | IEEE Standard 323 as opposed to other instruments reviewed recognizes the need for aging and defines steps to address aging such as age conditioning and natural aging criteria. |
| (28) | | 344 | Establishes recommended practices to obtain data to qualify that the equipment will perform one SEE after an OBE. Included are tests for vibrational aging, seismic aging plus normal operation loads. | Acceptance and qualifying records. | ? | None. | None. | |
| (29) | | 387 | Provides principal design criteria for EDGs to meet their functional requirements under design base conditions. | Provides an acceptable design base for possible LR. | Standard is being revised by IEEE working group 4.2. | None. | None. | Provides the minimum service requirements for the design staff (A&E) including operational cycles, operational hours, temperatures (max. and min.), seismic response, radiation, humidity, and air quality, i.e., all the conditions that should be considered in the design. Fortyseven (47) design and application considerations are given in the standard. |
| (30) | INTERNAL CHEMICAL PHYSICAL PROCESSES, AND CORROSION:<br>• CORROSION IN AIR START SYSTEM<br>• THERMAL SHOCK | SRPs 9.5.7 | EDG engine lube system - review determines if the lub system prevents deleterious material from thermal shock entering the lub oil system. | Records of acceptance criteria. | None. | None. | Needs acceptance criteria re-established. | |
| (31) | | 3.5.8 | EDG air intake system - review determines that no engine degradation will be experienced during maximum power output (continuous) settings. | Records of acceptance criteria. | ? | None. | Needs acceptance criteria re-established. | |

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|---|---|---|---|---|---|---|---|---|
| (32) | INTERNAL CHEMICAL PHYSICAL PROCESSES, AND CORROSION (contd) | ASME Code Section III Section XI | EDGs by definition are Class 1, 2, and 3 components; however, the EDGs are not pressure vessels or piping, etc. | None. | ? | None. | None. | The ASME codes as required in 10 CFR 50.55a mandate ISI. These requirements rarely involve EDG unless a specific part is designed and manufactured in accordance with the Code. |
| (33) | | IEEE 308 | Provides design base criteria for Class IE equipment to enable them to meet their functional requirements. | Provides an acceptable design base for possible LR. | ? | None. | None. | Includes power systems, e.g., diesel generators, design base includes malfunctions, accidents, or operating modes that could lead to degradation of the systems. Malfunctions are defined as natural phenomena, e.g., earthquakes and floods, and postulated phenomena, e.g., environmental factors of pressure, humidity, temperatures and accidents, e.g., fires. |
| (34) | | 387 | Provides principal design criteria for EDGs to meet their functional requirements under design base conditions. | Provides an acceptable design base for possible LR. | Standard is being revised by IEEE working group 4.2. | None. | None. | Provides the minimum service requirements for the design staff (A&E) including operational cycles, operational hours, temperatures (max. and min.), seismic response, radiation, humidity, air quality etc., i.e., all the conditions that should be considered in the design. Fortyseven (47) design and application considerations are given in the standard. |
| (35) | | 323 | Ability to perform safety function due to the effect of aging must be addressed. Types of aging include thermal conditions. | Acceptance and qualifying records. | ? | None. | None. | IEEE Standard 323 as opposed to other instruments reviewed recognizes the need for aging and defines steps to address aging such as age conditioning and natural aging criteria. |
| (36) | SURVEILLANCE: • INSPECTION • EXCESSIVE, HARSH AND FREQUENT TESTING • MALADJUSTMENT/ MISALIGNMENT • UNNECESSARY DISASSEMBLY FOR INSPECTION | TSs 3/4.8 | None. | None. | PNL staff through the NPAR program Technical Specification Aging Task are addressing the TSs for aging management. | The TSs should consider aging mechanisms/aging issues. | Provide evidence that the TSs are adequate for license renewal, i.e., are changes needed in the TSs for LR? | Describes minimal AC electrical power requirements for operating and shutting down plants. |
| (37) | | 3/4.8.1 operating status and 3/4.8.1 shutdown status | Surveillance efforts would detect an inoperable generator by virtue of the necessity to assure an operable generator; management of aging is implied. | Inspection records. | None. | Not applicable. | Records of maintenance inspection could provide evidence for extended use of the diesel generator. | |

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|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | SRPs | | | | | | |
| (38) | | 8.1 Appendix 8-A BTP ICSB(PSB)-8 | Use of generator for peaking power is not allowed; prevents use of generator in nonemergency situations. | Should provide record of use, i.e., total service hours. | ? | N/A | N/A | |
| (39) | | 8.3.1 | Provides review and acceptance criteria for operation of stand-by generator sets, especially as related operation at less than full load with no degradation for extended periods and that no load conditions shall be minimized. Stipulates that staff training, testing, preventative maintenance and repair procedures will be maintained. | Provides records for acceptance of stand-by power supplies. | None. | Is the acceptance criteria useful for long-term assurance that the EDG will be reliable, not degrade over time. | For LR, the required maintenance and training must be shown to be adequate for requalification of the EDGs. | The SRP does not specifically address aging or LR; however, strongly implies aging management and LR because the SRP states the EDG shall not be subjected to degradation. This may be assumed to cover the life of plant and beyond. |
| (40) | | 8.3.1 BTP-8 (PSB) | Provides restrictions for using the EDG for peak power situations and restricts overuse of the EDG. | Limits degradation of equipment and provides for a longer and useful life. | None. | None. | For LR, the required operating restrictions must be shown to be useful for life extension. | The BTP provides aging management by specifically restricting the use for peak power needs and controlling nonemergency use of the EDG. The BTP addresses the prevention of common failure mode of the EDG as related to main or off-site power. |
| (41) | | 9.5.5 | EDG engine cooling water system - review determines the adequacy of inspection and testing of the cooling water system. | Should provide adequacy records and status for cooling water system relative to LR. | ? | None. | None. | Although the SRP does not explicitly discuss aging management or LR, the requirements should support LR if adequate records are maintained throughout the life of the plant. |
| (42) | | 9.5.6 | EDG starting system - review determines adequacy of the inspection and testing of starting system. | Should provide adequacy records and status for starting system. | ? | None. | None. | Although the SRP does not explicitly discuss aging management or LR, the requirements should support LR if adequate records are maintained throughout the life of the plant. |
| (43) | | 9.5.4 | EDG fuel oil system - review determines adequacy of inspection and testing of fuel oil system. | Should provide adequacy records and status for fuel oil systems. | ? | None. | None. | Although the SRP does not explicitly discuss aging management or LR, the requirements should support LR if adequate records are maintained throughout the life of the plant. |
| (44) | | 9.5.7 | EDG engine lub system - review determines the adequacy of the inspection and testing of engine lub system. | Should provide adequacy records and status for engine lub system. | ? | None. | None. | Although the SRP does not explicitly discuss aging management or LR, the requirements should support LR if adequate records are maintained throughout the life of the plant. |

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|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | SRPs (contd) | | | | | | |
| (45) | | 9.5.8 | EDG engine combustion - review determines the adequacy of the inspection and testing of engine combustion system. | Should provide adequacy records and status for the engine combustion system. | ? | None. | None. | Although the SRP does not explicitly discuss aging management or LR, the requirements should support LR if adequate records are maintained throughout the life of the plant. |
| (46) | | 13.2.1 | Operators and other nonlicensed plant staff training - the plant staff are to be trained to use the EDGs correctly. This implies aging management. | Should provide plant staff performance training records, i.e., adequacy of current staff. | ? | None. | Are the current records up to date for present staff for adequate LR documentation? | The plant staff are expected to know how the EDG operates and performs; furthermore, they are expected to know how the EDG interacts within (other) plant safety-related systems. Documentation of training and demonstrations of trained staff could enhance aging management and LR applications. |
| | | RGs | | | | | | |
| (47) | | 1.9 | Provides qualification and periodic testing, test schedule and surveillance and maintenance guidance. | Provides records and records - keeping criteria and reporting criteria. | May be impacted by NPAR EDG studies by PNL. The NRC has issued 1.9 Rev. 3 for comments, November 1988. | Modification of fast start testing, start-run-cooldown requirements are needed to reduce aging stressors. | Current status/condition of the equipment should be addressed for LR. | RG 1.9 references 10 CFR 50, Appendix A, Criterions 17 and 18, and Appendix B, Criterion XI. |
| (48) | | Safety Guide Number 30 | Establishes regulations for QA relative to testing of electrical equipment. References ANSI N45.2 and IEEE-336. | QA documentation. | ? | ? | Status of records and recordkeeping method need to be verified for LR. | Safety guide reference 10 CFR 50, Appendix B, QA criterion for NPPs. |
| (49) | | 1.33 | Refers to overall general criterion for NPP operation. | Implied by documentation. | ? | N/A | A need exists to determine if the exiting records and QA documentation are adequate for LR. | |
| (50) | | 1.41 | Provides guidance for testing after a major modification. | Provides records history of new modifications. | ? | ? | ? | |
| (51) | | 1.58 | Establishes guidance for the qualification of NDE staff who perform inspections, tests and examination during the operations of NPPs. | Provides documentation of staff qualification and QA records, i.e., what is the status of past NDE performance? | ? | | | Refers to ANSI N45.2.6-78. |
| (52) | | 1.108 | Establishes that the EDG design should allow testing. The RG also defines the periodic preoperational 18-month testing requirements. | Establishes guidance for records and recordkeeping. | The NRC intends to delete RG 1.108. Guidelines will be in revised RG 1.9. | Testing should be changed to a slow start to avoid fast-start stressors. | Improved and less stressful starting procedures accompanied by a program of performance monitoring and trending are needed. | See "Recommended Practice for Aging Mitigation and Improved Programs for Nuclear Service Diesel Generators," NUREG-CR-5057/PNL-6309. |

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|---|---|---|---|---|---|---|---|---|
| (53) | SURVEILLANCE (contd) | ASME Code Section III Section XI | EDGs by definition are Class 1, 2, and 3 components; however, the EDGs are not pressure vessels or piping, etc. | None. | None. | None. | None. | The ASME codes as required in 10 CFR 50.55a mandate ISI. These requirements rarely involve EDG unless a specific part is designed and manufactured in accordance with the Code. |
| (54) | | GSIs GSI B-56 | Program to improve the reliability of EDG; goal of 95% success established. | ? | Diesel reliability program with NUMARC. | Reliability of EDG as a function of age; what are effects of age, effects of testing over time? | Effects of reliability decrease on licensing. | See also Regulatory Guide 1.108; NUREG/CR-0660. |
| (55) | | GSI-91 | Continued reliability and operability of Transamerica DeLaval, Inc., EDGs. | ? | Transamerica DeLaval, Inc., Owners Group. | Effects of stress, fatigue, testing on TDI EDG reliability - do TDI EDG show aging effects faster than other models? | Can plants with TDI EDG continue to operate; how to verify operability of TDI EDG after thirty years of service? | Issue involves potential main crankshaft failures. See also SECY-83-84, IEN-83-58, and NUREG-1216 and other models? |
| (56) | | ANS 3.1 | Provides criteria for selection, qualification, and training of personnel for stationary nuclear power plants. | Provides record or training of staff. | ? | None. | None. | |
| (57) | | 3.2 | Provides recommendations and requirements for administrative controls, including written procedures, and QA program to help assure that activities of NPPs are carried out without risk to the health and safety of the public. | Administrative in-place controls will probably be required for LR. The associated QA documentation will be useful for LR. | ? | None. | None. | Among the activities covered under this standard are design changes, fabricating, cleaning, decon, inspecting, testing, maintaining, and repairing. |
| (58) | | IEEE 338 | Provides design and operational criteria for periodic testing of safety systems. | Acceptance testing of components for LR. | ? | None. | None. | This standard does not address maintenance. |
| (59) | | 336 | Provides minimum requirements for inspection and testing of Class IE power, instrumentation, and control equipment during construction phase. | Records of construction may be applicable to LR. | ? | None. | None. | Provides a criteria for correct installation which could effect aging aspects of the equipment after start-up and operation. Class IE is equipment that is essential to emergency shutdown, containment isolation, RCC, and CHR. |

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|---|---|---|---|---|---|---|---|---|
| | SURVEILLANCE (contd) | IEEE (contd) | | | | | | |
| (60) | | 749 | Provides standard for periodic testing of EDG including availability tests, system operational tests, and independence verification tests. | Records of past performance. | Standard will be impacted by RG 1.9. IEEE plans to delete this standard. | Less stressful starting procedures and tests. | Less stressful starting procedures and test. | |
| (61) | | 934 | Provides criteria for replacement parts, both construction and operation. Requires inspection and test of parts prior to release for service. | Acceptance testing and records that may be applicable to LR, e.g., failure review for wear, fatigue, original defects, and insulation breakdowns. | None. | None. | None. | Standard determines selection of required parts for Class IE equipment, e.g., identification and failure review. However, required design changes are outside the scope of this standard. |

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11. ABSTRACT (200 words or less)

This report comprises Part I of a review of U.S. nuclear plant regulatory instruments to determine the amount and kind of information they contain on managing the aging of safety-related components in U.S. nuclear power plants. The review was conducted for the U.S. Nuclear Regulatory Commission (NRC) by the Pacific Northwest Laboratory under the NRC Nuclear Plant Aging Research (NPAR) Program. Eight selected regulatory instruments, e.g., NRC Regulatory Guides and the Code of Federal Regulations, were reviewed for safety-related information on five selected components: reactor pressure vessels, steam generators, primary piping, pressurizers, and emergency diesel generators. The focus of the review was on 26 NPAR-defined safety-related aging issues, including examination, inspection, and maintenance and repair; excessive/harsh testing; and irradiation embrittlement. The major conclusion of the review is that safety-related regulatory instruments do provide implicit guidance for aging management, but include little explicit guidance. A major recommendation is that the instruments be revised or augmented to explicitly address the management of aging.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Regulatory instruments, safety-related components, codes and standards, guides, general design criteria, life extension and life extension needs, aging management and aging issues including: corrosion, erosion, environment effects, harsh testing, wear, embrittlement, fatigue.

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